INDUSTRIAL RADIOACTIVE WASTE DISPOSAL

HEARINGS

BEFORE THE

SPECIAL SUBCOMMITTEE ON RADIATION

OF THE

JOINT COMMITTEE ON ATOMIC ENERGY

CONGRESS OF THE UNITED STATES

EIGHTY-SIXTH CONGRESS

FIRST SESSION

ON

INDUSTRIAL RADIOACTIVE WASTE DISPOSAL

JANUARY 28, 29, AND 30; FEBRUARY 2 AND 3, 1959

Volume 1

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NOTE

This is Volume 1 of four volumes of the record of "Hearings on Industrial Radioactive Waste Disposal." It covers the portion of the hearings concerned with the origin and nature of various types of radioactive wastes evolved from nuclear energy activities and operations in use to manage these wastes at various AEC and other installations. Volume 2 deals with the first half of the current waste disposal research and development program. It describes and discusses the treatment and controlled dispersal of low and intermediate level wastes to the atmosphere, surface waterways, the ground, and the ocean and environmental investigations associated with this phase of the development program. Volume 3 covers the second half of the current waste disposal research and development program. It describes and discusses the handling, treatment, and proposed ultimate disposal systems for highly radioactive wastes that are primarily evolved from irradiated fuel reprocessing plants and discusses the future estimates and economics of disposal of radioactive wastes. It also includes a summary of environmental factors related to these estimates and a discussion of the industrial viewpoint of the radioactive waste problem. Volume 4 discusses the activities of Federal and State governmental agencies in radioactive effluent control and administrative relationships between these agencies. The international aspects of the problem are also dealt with in this volume.

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INDUSTRIAL RADIOACTIVE WASTE DISPOSAL

WEDNESDAY, JANUARY 28, 1959

CONGRESS OF THE UNITED STATES,
SPECIAL SUBCOMMITTEE ON RADIATION,
JOINT COMMITTEE ON ATOMIC ENERGY,
Washington, D.C.

The special subcommittee of the Joint Committee on Atomic Energy met at 10 a.m., pursuant to notice, in the old Supreme Court chamber, the Capitol, Hon. Chet Holifield, chairman of the special subcommittee, presiding.

Present: Representatives Holifield (presiding), Durham, Price, Van Zandt, Hosmer, and Bates, and Senator Dworshak.

Present also: James T. Ramey, executive director, David R. Toll, staff counsel, and Dr. Walton Rodger, technical adviser, of the Joint Committee on Atomic Energy.

Representative HOLIFIELD. These public hearings on Industrial Radioactive Waste Disposal are a part of a series of hearings on radiation hazards begun by the Joint Committee in 1957. Hearings have already been held on reactor hazards, and the radioactive fallout problem from weapons tests.

These hearings will consider questions of great importance to the public and to the future safe development of the world's atomic power industry. Some of these questions can be of local importance in communities where atomic energy activities are centered, but others are also international in scope, in trying to establish common standards and safeguards.

The subcommittee and staff became particularly interested in the problem of waste disposal during the hearing in 1957 on fallout from weapons tests. Several of the experts who testified stated that the hazards of radioactive wastes from the peacetime atomic power industry will be far greater than fallout from the weapons tests.

While radioactive wastes from atomic reactors are obviously much more subject to control than fallout from weapons tests once released, we want to see what the nature and magnitude of the production of radioactive wastes will be, and how good the methods of waste disposal are and will be.

Indeed, the waste disposal problem may be just as neglected from a public standpoint, as fallout was in 1954. I hope that we will not require a series of incidents such as the contamination of the Japanese fishing boat, the *Lucky Dragon*, from the Marshall Islands test, to focus public attention on the waste disposal problem.

We hope in these hearings to bring out in a thoughtful and objective manner, the various facets of the waste disposal problem.

Considerable has been written on the subject, but it has been mostly in expertise. We will have experts in these hearings, but they have agreed to talk in laymen's terms.

Some idea of the magnitude of waste disposal problems can be gained by a quotation from a recent article by a scientific authority in this field, Dr. Wallace de Laguna, entitled "What Is Safe Waste Disposal." I quote:

> * * * By the year 2000, if we have in operation 700,000 megawatts of installed heat capacity, we will be destroying by fission and neutron capture several hundred tons a year of nuclear fuel (U^{235}, U^{233}, and Pu^{239}) and producing about 280 metric tons of fission products. This well very largely be contained in some 50 million gallons a year of high-level waste.

I might interpolate that 700,000 megawatts of heat capacity is approximately 200 million kilowatts of electricity.

Another question concerning waste products might be, How are they produced and how long will they persist as a hazard?

The article points out that one of these radioisotopes produced, americium 241, has a half life of 510 years, which means that the waste will still retain some radioactivity "even after several thousands of years."

The article estimates that the quantity of the single isotope of strontium 90 produced will be so large by the year 2000 that roughly 16 million cubic miles of water would be needed to safely dilute it. That's more fresh water than there is in the world, including the polar icecap.

The quantity of strontium 90 expected to exist by the end of the century, he says, shows that its potential hazard, unless contained, is more than enough to injure all mankind. This gives us at least one authoritative opinion on the scope of the problem we are facing.

I have quoted from this article because it is current and because it gives us at least an authoritative opinion on the scope of the problem we are facing.

However, the author himself cautions against the public assuming an alarmist's point of view. In these hearings we too will attempt to approach the problem objectively and fairly, without trying either to discount or magnify the potential hazards of radioactive waste products.

Other questions which we will study during these hearings are as follows:

What precautions has the Atomic Energy Commission taken to dispose of these wastes in its own activities?

What standards and safeguards have been suggested in connection with the licensing of new privately owned atomic powerplants?

What research and development work is being done by the Government and by industry to find safe means of disposal of these wastes, and is the level of support for these research and development efforts adequate?

What are the estimates of the future quantity of these wastes that will be generated as an atomic power industry is developed, and how does the disposal question affect the economics of our embryo atomic power industry?

What are the possibilities of developing beneficial uses of some of these waste products?

In this connection I note that the recently disclosed SNAP III radioactive battery could use a waste product rather than polonium which is more hazardous.

What steps are being taken by Government agencies to regulate radioactive wastes—on the Federal, State, and local levels?

Finally, since the United States is cooperating and assisting many other countries in an internatonal atoms for peace program, what efforts have been made to develop international standards and safeguards for the control of radioactive wastes?

Scheduled witnesses and approach of hearings:

In order to search intelligently for answers to the above questions, our subcommittee has invited some of the leading experts and minds in the country to contribute papers and discuss these problems with us. We do not wish to approach these problems with fear or prejudice, but to hear the opinions of persons who have worked in the field, who have experience in handling these wastes, who are working in the research and development program, or making plans for the future.

Today, for example, we are scheduled to hear from our distinguished and recognized leaders in the scientific community, Dr. Abel Wolman of Johns Hopkins University;

Dr. Floyd L. Culler of the Oak Ridge National Laboratory;

Dr. Herbert Parker of the Hanford Laboratroy, and

Dr. K. Z. Morgan, of the Oak Ridge National Laboratory.

In order to cover fully the many important subjects in the limited time available to the subcommittee, we have requested that each expert witness prepare a short oral statement, and that he keep within the time limit indicated in the outline.

We have asked that these statements be as nontechnical as possible because the hearings are designed to inform Congress and the public about this increasingly serious, but complex matter.

Each witness presenting oral testimony has been requested to supply additional technical data, including charts or graphs if he so desires, in order that the record of the hearings may contain all the necessary and available technical information.

Finally, we have also requested papers for the record from many leading experts throughout the country on subjects on which they have done work.

I might interpolate here to say that if there are any witnesses who are expert in this field and who have had a background of experience in the field who are not on the list of witnesses and cannot be put on because of the time limit involved, these people will have the opportunity of submitting papers to us which may be printed in the hearings or in the summary of the report.

The approach to be followed in general is the same as that adopted by the Special Subcommittee on Radiation during its first public hearings held in the spring of 1957 on "The Nature of Radioactive Fallout and its Effect on Man."

During those hearings we also conducted, whenever time permitted, informal round table discussion among the experts and subcommittee members at the close of each morning or afternoon's testimony. We hope to repeat that procedure during these hearings because we believe that the panels are helpful in developing new ideas and concepts and in identifying sources of agreement and disagreement.

The outline of the hearings and list of witnesses invited to participate was prepared by the committee staff with the valuable assistance of an informal advisory panel of experts. The record will be held open until February 15, 1959, for additional statements.

The committee staff also received excellent cooperation and assistance from the Atomic Energy Commission staff in preparing for these hearings.

Finally, the subcommittee has been fortunate in obtaining as a technical consultant to the subcommittee and its staff during these hearings, Dr. Walter Rodger of the Argonne National Laboratory, who has had considerable background experience in this field.

The outline and list of witnesses and a sheet of instructions, has been prepared and distributed in advance to all of the persons and organizations who have been requested to appear before the subcommittee or submit materials for the record.

At this point, I would like to insert in the record a copy of the outline and list of witnesses dated January 8, 1959, and also a copy of the sheet of instructions of the same date.

(The material referred to follows:)

JOINT COMMITTEE ON ATOMIC ENERGY,
CONGRESS OF THE UNITED STATES,
SPECIAL SUBCOMMITTEE ON RADIATION,
January 8, 1959.

OUTLINE AND LIST OF WITNESSES

PUBLIC HEARINGS ON INDUSTRIAL RADIOACTIVE WASTE DISPOSAL, JANUARY 28, 29, 30, AND FEBRUARY 2, 3 1959, ROOM P-63, U.S. CAPITOL (OLD SUPREME COURT CHAMBER)

Notes.—1. All hearings will be held in public and members of the public are invited. Room P-63 of the Capitol (the old Supreme Court Chamber) has been reserved for the hearings on January 28, 29, February 2 and 3, but the hearings on Friday, January 30, may have to be held in another hearing room. The hearing room for January 30 will be announced on January 28.

2. The hearings for all days except Thursday, January 29, have been scheduled for the a.m. only, (commencing at 10 a.m. each day), but will be continued in the p.m. if necessary or desirable. On Thursday, January 29, (the second day) the hearings have been scheduled for both the a.m. and p.m.

3. In addition to the papers listed in the outline below, others may be requested or considered for inclusion in the record of the hearings.

FIRST DAY (WEDNESDAY, JANUARY 28)

I. Introduction

General statement by a leading authority on nature and extent of waste disposal problems: Dr. Abel Wolman, Johns Hopkins University, 20 minutes.

II. Nature of wastes

Will cover sources of waste from nuclear energy operations, the quantities of waste generated, and the waste characteristics, including physical, chemical, and radioactive composition, with emphasis upon the significance of these constituents. Wastes in the solid, liquid, and gaseous phases should be discussed. Discussion of the management of these wastes is not to be included in this section.

A. F. L. Culler, ORNL, 40 minutes.

B. Papers for the record have been requested from—

1. Ore mining and processing (Raw Materials Division, AEC).
2. Fuel element manufacture (N. C. Jessen, Babcock & Wilcox Co.).
3. Reactor operations (Argonne National Laboratory).
4. Reactor fuel reprocessing (Hanford, Idaho, ORNL).
5. Isotope uses (L. Rogers, Division of Licensing and Regulation, AEC).

III. Waste-management operations

Witnesses from individual areas or installations will relate in detail existing operational methods and procedures utilized in the collection, handling, processing, and disposal of waste materials. Development work will be handled primarily under IV and should be discussed here only as it pertains to the establishment of present operating practice. The environmental and economic considerations in these operations should be included.

A. Hanford: H. Parker, 30 minutes.

B. Oak Ridge: K. Z. Morgan, 30 minutes.

SECOND DAY (THURSDAY, JANUARY 29) A.M.

C. Idaho: J. R. Horan (IDO) and R. L. Doan (Phillips Petroleum Co.), 30 minutes.

D. Shippingport pressurized water reactor (PWR): A. L. Bethel, Westinghouse Co., 15 minutes.

E. Papers for the record have been requested from: AEC, Division of Raw Materials; Tracerlab, Inc.; Argonne; Brookhaven; Knolls Atomic Power Laboratory; Los Alamos; Savannah River; the University of California Radiation Laboratory; Army Package Power Reactor; Nuclear Aircraft Research Facility; Convair; and Naval Reactors Program.

IV. Waste-disposal research and development program

Witnesses will testify on research and development work which has been or is being performed for the AEC on the disposition of radioactive waste material. Significant results of completed work and objectives of present areas of study should be reported.

A. Introduction (J. A. Lieberman, AEC), 15 minutes.

B. Low and intermediate level wastes:

1. Surface waterways: Harold A. Thomas, Harvard, 15 minutes; R. F. Foster, Hanford, 15 minutes; E. C. Tsivoglou, USPHS, Cincinnati (paper for the record); AEC Division of Raw Materials (paper for the record).

2. Ground disposal: C. V. Theis, USGS, Albuquerque, N. Mex., 15 minutes; D. W. Pearce, Hanford (paper for the record); K. E. Cowser, ORNL (paper for the record).

3. Ocean and tidal estuary disposal: D. W. Pritchard, Chesapeake Bay Institute, 15 minutes; R. Revelle, Scripps Institute of Oceanography (paper for the record).

SECOND DAY (THURSDAY, JANUARY 29) P.M.

4. Meteorologic studies: D. Pack, U.S. Weather Bureau, 15 minutes; G. R. Hilst, Hanford, 15 minutes.

5. Air cleaning: L. Silverman, Harvard, 15 minutes.

THIRD DAY (FRIDAY, JANUARY 30)

C. High-level wastes:

1. Fixation and ultimate disposal: L. P. Hatch, Brookhaven, 15 minutes; E. G. Struxness, ORNL, 10 minutes; R. L. Doan, Phillips Petroleum Co. (Idaho chemical processing plant), 15 minutes; Henry C. Thomas, University of North Carolina (paper for the record).

2. Direct disposal: W. B. Heroy, the Geotechnical Corp., Dallas, Tex., 15 minutes; T. V. Moore, Standard Oil Co. of New Jersey, 15 minutes; E. G. Struxness, ORNL, 10 minutes.

3. Removal of specific isotopes: R. E. Tomlinson, Hanford, 15 minutes; R. E. Blanco and J. T. Roberts, ORNL (paper for the record).

D. Fission product utilization: Discussion on the feasibility of removing and utilizing fission products from waste streams for beneficial use. Included would be information on an estimated market potential for these products and the effect on future waste processing and disposal requirements.

1. B. Manowitz, Brookhaven National Laboratory, 15 minutes.

2. A. F. Rupp, ORNL, 15 minutes.

3. AEC Office of Isotope Development (paper for the record).

FOURTH DAY (MONDAY, FEBRUARY 2)

V. Future estimates and economics of waste disposal

A. Discussion on future estimates of waste quantities, as related to storage, processing, and ultimate disposal procedures, plant locations, transportation and

costs and the effect on nuclear power development: Frank R. Bruce, ORNL, 30 minutes.

B. Summary and evaluation of environmental factors: Would include information and discussion on the possible effects of future quantities of radioactive wastes on man and his environment: H. Parker, Hanford, 15 minutes; K. Z. Morgan, ORNL, 15 minutes; J. G. Terrill, U.S. Public Health Service, 15 minutes; A. Wolman, 15 minutes; R. Revelle, Scripps Institute, 15 minutes.

C. Industrial viewpoint:

1. P. Sporn, 20 minutes.

2. Papers for the record have been resquested from American Standards Association; AEC Division of Reactor Development; Commonwealth Edison Co.; General Electric Co., Vallecitos.

D. Roundtable discussion, 15 to 30 minutes

FIFTH DAY (TUESDAY, FEBRUARY 3)

VI. Activities of Federal and State agencies in regulating disposal of radioactive wastes

A. AEC (H. Price, Division of Licensing and Regulation), 20 minutes.

B. U.S. Public Health Service (D. E. Price), 20 minutes. Discussion to include the PHS activities in stream pollution abatement and matters involving the radiological health and safety of people and their environment.

C. State agencies:

1. C. W. Klassen, Illinois State Department of Health, 20 minutes.

2. H. S. Ingraham, New York State Department of Health, 15 minutes.

3. Papers for the record as included in D below.

Discussion would center on the State agencies' relationships with AEC and AEC contractors in the construction of nuclear facilities and the use of radioactive materials in laboratories, hospitals, etc., with emphasis on waste handling problems.

D. Papers for the record have been requested from other Federal and State agencies, including U.S. Weather Bureau, USGS, ICC, FDA, Maritime Administration, Ohio River Valley Water Sanitation Commission, and States of California, Colorado, Connecticut, New Mexico, Ohio, Pennsylvania, Texas, and Washington.

VII. International aspects

1. A. E. Gorman, Geneva Conference on International Law of the Sea, 15 minutes.

2. D. W. Pritchard, IAEA waste disposal activities, 15 minutes.

3. L. S. Taylor, International Commission on Radiological Protection, 15 minutes.

VIII. Roundtable discussion

Witnesses from VI and VII, 15 to 30 minutes.

INSTRUCTIONS

1. *Papers for the record, 3 copies required.*—Papers for the record may be as long as necessary to cover the subject adequately, and may contain detailed technical data.

2. *Oral testimony, 50 copies required.*—Because of limitations of time, oral testimony must be limited to the time indicated in the outline. The oral statements should be summaries, short and understandable to the lay public. The use of visual aids in presenting oral testimony is encouraged. Additional material, particularly technical data, charts, graphs, diagrams, photographs, etc., should be submitted for the record by those presenting oral summaries.

3. *Use of graphic material and visual aids.*—Where charts or photographs are submitted for the record, they should be black and white glossy prints. Since the record will be printed in black and white, if colored charts are used, the text should refer to shades such as "light," "dark," etc., rather than by the color. If large charts, etc., are used in connection with oral testimony, at least 3 smaller copies should be submitted for the record. Slide projection facilities are not available unless provided by the witness.

4. *Date for submission of material to Joint Committee.*—Papers for the record—January 27, 1959. Oral statements—please deliver to Mr. Toll, of the Joint Committee staff, as soon as practicable after your arrival in Washington, D.C. for the hearings, not later than 10 a.m. of the day of your appearance before the subcommittee.

5. *Biographical sketch.*—Three copies of a short biographic sketch of all persons presenting statements (oral or written for the record) are requested. Such sketches should include: name; date and place of birth; education; work history.

6. *Room of hearings.*—P–63 of the Capitol (Old Supreme Court Chambers) for January 28, 29, February 2 and 3. Hearing room for January 30 will be announced on January 28.

7. *Further information.*—If you have any further questions, please contact either the Joint Committee staff (James T. Ramey, executive director or David R. Toll, staff counsel) or the AEC staff (Dr. J. A. Lieberman or Walter Belter, Division of Reactor Development).

Representative HOLIFIELD. Our first witness this morning is Dr. Abel Wolman, of Johns Hopkins University, a member of the Advisory Committee on Reactor Safeguards, a distinguished scientist and educator, and a recognized authority in the field of radioactive waste products.

We are glad to have you with us this morning, Dr. Wolman. Please proceed.

STATEMENT OF DR. ABEL WOLMAN, THE JOHNS HOPKINS UNIVERSITY [1]

Dr. WOLMAN. Mr. Chairman, members of the committee, I have filed with the committee a statement in considerable detail and I shall take the liberty of commenting upon various portions of it as I go along.

---

[1] Place of birth: Baltimore, Md. Date of birth: June 10, 1892. Bachelor of arts, 1913; bachelor of science in engineering, 1915; honorary degree, doctor of engineering, the Johns Hopkins University, 1937.
1921 to 1937: Editor in chief, Journal of the American Water Works Association.
1922 to 1939: Chief engineer, Maryland State Department of Health.
1929 to 1930: Consulting sanitary expert to State of New Jersey in U.S. Supreme Court case re *Delaware River, New Jersey* v. *New York, N.Y.*
1931 to date: Consulting engineer on water supply, sewerage, refuse disposal, etc., Baltimore City Department of Public Works.
1934 to 1937: Acting State director, Federal Emergency Administration of Public Works (Maryland and Delaware).
1937 to date: Professor of sanitary engineering, the Johns Hopkins University.
1937 to 1944: Consulting engineer, U.S. Public Health Service, U.S. Engineer Corps investigations of the Ohio River pollution.
1938, 1945, 1949, 1956: Consulting engineer, Washington Suburban Sanitary District.
1939 to 1945: Consultant, U.S. Public Health Service.
1939 to date: Consulting engineer, Tennessee Valley Authority.
1944 to date: Consultant, Surgeon General, U.S. Army.
1934 to 1945: Chairman, Maryland State Planning Commission.
1939: President, American Public Health Association.
1942: President, American Water Works Association.
1942 to date: Chairman, Advisory Committee on Sanitary Engineering and Environment, Division of Medical Sciences, National Research Council, Washington, D.C.
1942 to date: Chairman, Permanent Sanitary Engineering Committee, Pan American Sanitary Bureau.
1943 to 1946: Chairman, Interdepartmental Board for Sanitary Control and Protection of Public Water Supply of New York City.
1945 to date: Chairman, board of consultants, Jordan River project, Israel.
1947 to date: Consultant, Atomic Energy Commission (ACRS, Stack Gas Working Group, general wastes problems).
1949 to 1952: Chairman, Expert Committee on Environmental Sanitation, World Health Organization.
1955: Editor, American Journal of Public Health.
1955: Consultant, National Science Foundation.
1955: Consultant, National Defense Department.
1955: Scientific adviser, U.S. delegation, International Conference on Peaceful Uses of Atomic Energy.
1955: Consultant, Government of Ceylon, water resources.
1956 to 1957: Member, International Mission, Organization of Government of Argentina.
1956 to 1957: Member, board of consultants, Detroit Metropolitan Region.
1956 to 1957: Member, board of consultants, Miami Conservancy District.
1957: Member, board of consultants, Indianapolis Water Co.
1957: Member, National Advisory Committee on Radiation to the U.S. Public Health Service.
1958: Advisory Committee, USPHS, Washington, D.C.
1958: Radiation Control Committee, Johns Hopkins Medical Institutes.
1959: Special Advisory Committee, Department of Commerce, Washington, D.C.

It is rare, I believe, to be able to discuss the radioactive waste problems of an industry as new and as recent as the atomic energy industry.

I recall to the committee that its history is approximately 15 years in length. In our own country it now represents something in excess of $6 billion in investments.

This atomic energy industry through the control which has been primarily exercised by the Atomic Energy Commission has so far kept the hazards of these wastes to a level well below that which would endanger either the worker or the general population.

It is not extraordinary that any evolving industry produces wastes. This is a familiar part of the entire industrial revolution. Every industry of which I am aware produces waste which in turn has to be disposed of by either an artificial process or by release to nature.

This particular industry is not peculiar in the sense of producing waste. It is, however, peculiar in producing a series of wastes which up to the present is not disposable in the commonsense or familiar meaning of the word "disposed." We dispose of the wastes of almost every industry in the United States by actual conversion into harmless materials, generally in nature, either into surface waters, into the air, or into the ground.

This is the first series of wastes of any industry where that kind of disposal is nonexistent in the present state of knowledge and of the art.

For the purposes of this hearing and the setting in which subsequent witnesses will speak, it should be pointed out that we have a wide variation in the nature and characteristics of these materials. There are great differences in concentration in the kind of radioactive materials in the gaseous, liquid, and solid states.

The volumes to be managed differ according to the processes and their physical and chemical characteristics have similar wide variations.

Up to the present time the managements of these radioactive waste materials has followed two general precepts. I want to stress those principles because it is a point of departure with respect to all subsequent discussions. These precepts, although not stated officially by the Atomic Energy Commission, are generally as follows:

With a high-level radioactive waste the general policy has been to concentrate wherever possible and to contain.

In more familiar terms, to reduce their volume if you can and to hold the material. This confirms my earlier comment that none of these has been disposed to nature in the common, familiar, orthodox sense.

With low and intermediate level wastes the precept has been to dilute wherever possible and to disperse to nature, either into the air, into waters, or into the soil, always, of course, with complete monitoring.

The utilization of fissionable material in the controlled production of energy, of course, causes the production of radioactive waste at each step, from the mining of original materials to the ultimate point of fission within a reactor or an equivalent facility.

The industry produces waste at each stage and it is at each stage that these wastes must be considered, controlled, supervised, and adjusted either for holding or for release.

The levels of activity in general are quite low in the earlier stages of all of these processes from the mine to the facility and reach their peak in the fission process and in the subsequent handling of materials resulting from the fission process.

There are two characteristics of these wastes that differ from any known familiar industrial waste hitherto managed by society. One is that these wastes by and large are not detected by the human senses.

In other words, they are not subject to human taste or odor. They give no indication to the average individual as to their existence.

The second characteristic of significance to society is that their toxicity in general terms, both radioactive and chemical, is greater by far than any industrial material with which we have hitherto dealt in this or in any other country.

The half life of the materials varies, of course, as always, from matters of seconds to many hundreds of years.

Representative HOLIFIELD. You might stop there and explain for the record the term half life in layman's language.

Dr. WOLMAN. These materials have a capacity for deterioration or for aging which is related to time. When we speak of a half life we speak of that period of time in which they may have lost their original capacity for radiation, not necessarily their toxicity, but they have lost about half of their radiation in a given time.

The reason for stating it in such form is that the radiation never reduces itself to exactly zero over the complete lifetime of a material, so that the scientists have chosen a half life which simply indicates the period in which it loses half of its capacity.

It then burdens us, of course, with an estimation of how much is left over subsequent years or subsequent time with which we have to deal.

The significance of the half life, particularly in the industrial waste field where we are dealing with radioactive materials which may have a half-life radiation of hundreds of years, is that here again we are confronted with a problem of supervision and control of a guaranteed nature which is unlike anything else that we have ever encountered.

In other words, we have to have continuity of governmental supervision, whether long or short, whether strong or weak. This is not a problem, in other words, which can be tackled from the standpoint of temporary expedience.

It is a problem which will require a deep governmental supervision, a very long continued and uninterrupted supervision over the fate and the location of these materials.

Representative HOLIFIELD. This means that this has to be taken care of for hundreds of years into the future?

Dr. WOLMAN. There is no way of which I can conceive in which these could be simply forgotten.

Representative HOLIFIELD. So it is very important that the policy decisions on the methods of storing these wastes permanently, you might say, be made wisely at this time because, if they are not made wisely, the hazard may erupt a hundred, 500, or a thousand years from now?

Dr. WOLMAN. That is correct. This cannot be left to chance.

One of the reasons for the primary policy which the Commission has exercised up to date; namely, of holding, rests on the assumption

that it is better to have this highly active waste in a situation where you know where it is, where you know what to do with it, where you do not leave it to chance for any given length of time in the future.

Representative HOLIFIELD. So it would be accurate to say that the problem of permanent disposal of high-level waste has not been solved; that it is in a state of suspension; that we are holding these high-level wastes to the extent of many millions of gallons in a temporary custody type of control and that no decisions have been made as to the final disposal of the high-level wastes.

Dr. WOLMAN. Unless there is some disclosure this week before this committee, which would modify your statement, I would say that that is correct.

Representative VAN ZANDT. Is it not true, Doctor, they have to have experience before they can make any firm decisions?

Dr. WOLMAN. Well, experience in this sense is essential before any firm decision can be made. It is an experience in research and development. It is not a mere experience in the creation of additional facilities for power production or for testing or for research.

What is lacking, as I am sure will be spelled out during the week, is the complete knowledge which will lead us to a variety of solutions not yet apparent.

In the absence of other solutions, holding is the principal one. I speak here as you do with respect to high level radioactive materials.

Representative HOLIFIELD. Of course, the very length of life of this hazard precludes us from having a history of operational experience. The experience that must be developed, as you say, is an experience in the scientific field of projection as to the effect on containers and storage areas and so forth, of the deterioration of the types of material which we use to contain these materials.

Whether it be in the field of stainless steel or in the field of concrete tanks, or whatever it may be, there has to be a projection and an estimation of the probable deterioration of these containers in order for us to project any kind of reliable estimate as to the wisdom of our choice of today in handling these materials.

Dr. WOLMAN. I should add, however, that choices are being made every day. Modifications are being made every day not only in choice of materials, but even in nature and process as the research indicates.

There is, however, another route which may modify some solution, namely, in going back into the initial fission process which creates these problems and so modifying fuel elements and fuel element processing so that the concentration, the level of radioactivity and so on of the waste may be lowered. That is the second route by which this may be done.

I may say, even though I would like to repeat it a little bit later on, all of this has more than theoretical interest, even in the protection of environment in general. It has a practical interest in relation to the evolution of power development from nuclear fission because this represents not only in absolute numbers, but in relative numbers a fairly high economic cost.

To that extent it has tremendous interest to everyone in the field.

Is there any way in which those relative costs can be reduced so that their load on power production costs in turn are reduced?

Representative HOLIFIELD. Is it not true that the disposal of the wastes, which have been generated and which are about to be generated by power reactors and other types of reactors throughout the country, has never been reduced to an economic charge against the cost of power production?

Dr. WOLMAN. As far as I am aware, that is still a heavy load which restrains the economy and the technology of power production to an extent that is undesirable for the long pull.

Representative HOLIFIELD. With respect to the various power reactors which have been made there are still factors of unknown costs which have not yet been attributed to the actual cost of construction and operation of the reactors. This is an area in which the Government has assumed and in which the Government in the main is assuming the financial responsibility.

Dr. WOLMAN. This is undoubtedly true and it is a rather interesting, if subtle, observation that in conversation with industrialists interested in nuclear fission power, they consider the waste problem to be quite unimportant, I believe for the psychological reason which you state.

It is unimportant to them because they are not responsible for its management and hence its cost.

Representative HOLIFIELD. That is right, and the method of handling it, to paraphrase, has been to ship it to Oak Ridge or some other place and forget about it.

Dr. WOLMAN. Under the provisions of the act, as you state, the Atomic Energy Commission up to the present time is responsible for taking care particularly of high level radioactive materials and, therefore, it disappears almost from the account book of the power developer.

Representative HOLIFIELD. But it does not disappear from the taxpayers' liability?

Dr. WOLMAN. It does not disappear from my own account book, either, in technology or in economics.

Representative HOLIFIELD. Or in responsibility for the hazard.

Dr. WOLMAN. Yes, without question it is not a sleight of hand operation.

The wastes are there regardless of who manages them.

Representative VAN ZANDT. At this point, and I am talking now about the waste material on hand, in your opinion has there been a laxity on the part of the Commission in this working out a program to properly handle this waste material and also protect the people from its effects?

Dr. WOLMAN. No, I should think the Commission ought to be congratulated on the efficiency with which it has so far managed this difficult operation.

If I were to make any comment at all it would be that the amount of energy, money, and personnel that ought to be devoted to this, from my own point of view, which may be a prejudiced one, is too low.

Representative HOLIFIELD. You are applying that to the field of research and development and the origination or development of methods to handle rather than to the factor of safety, of containment, with which we are quite well satisfied?

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Dr. WOLMAN. That is right. I would say that safety has been amply protected, the Commission has done a superb job in managing a hitherto unknown, unfamiliar, and peculiar material, and it has done it well.

As I pointed out, perhaps too briefly, it has done it without hazard to either the worker or to the general public.

What I do emphasize is that as time goes on and as the area and as the art develop, as we all hope it will, this problem will become more and more plaguing. Solutions to it will demand more time, energy and money.

Representative VAN ZANDT. Doctor, what you are saying now is that had the Commission started on a research program concerning the disposal of waste years ago we would be further ahead today than we are?

Dr. WOLMAN. Let me say this: I think, and this I am saying without detailed checking of the history of this program, I would say the Commission started research and development in the disposal of radioactive waste almost from the very beginning. Our difference, if any, might be entirely as to the amount of money devoted to it, the rate at which it might be expended and the pressure to get some solutions.

Over the 15-year period there has also been an evolution in understanding and in point of view. There was a period, perhaps 10 years ago, when the problem of radioactive waste was considered to be nonexistent.

It is not a judgment that I shared at any time, but it took a good deal of time before many people interested in the total nuclear fission field were willing to acknowledge the existence of an expanding difficulty in radioactive waste.

There, again, I would say that is exactly analogous to what happens in any industry. Without picking out any of our familiar industries, I would say, from my dealings with them over the years, when first created they will announce there are no wastes and before very long they will discover there are and still later on they will discover that something has to be done with them and even later on something is done with them.

This is a slow process. I would say the rapidity with which it has been recognized, managed and controlled by the Atomic Energy Commission is extraordinary.

Representative HOLIFIELD. Our records show that Dr. Rodger, our consultant, wrote his first paper on this in 1943. Of course, at that time there was very little of this to be dealt with, but as the years have gone along and as these reactors have multiplied both from the weapons standpoint and the now encroaching power development reactor program, the problem becomes larger and larger, the gallonage of this material becomes greater and greater and the concentration of toxicity becomes greater.

The public should know that the Atomic Energy Commission has been alerted to this problem and has been very diligent, in my opinion, in handling it on the only basis they know how to handle it, which is a temporary method of containment and custody.

They also should know this is a field where a permanent solution has not been found.

Dr. WOLMAN. I think that is a good summary of the situation. There has been an awareness leading to a very rigid control of the disposal or the management.

Secondly, an awareness has existed that this is not the final solution and that something must be pressed in research and development to arrive at a better solution.

I am not sure that there is a final solution. As far as I am aware there is no country in the world in which a solution for the kinds of high level radioactive wastes that we are concerned with has been found.

I say this with some reasonable knowledge of the practice in England, in Holland, in France, and to a lesser extent even in Russia.

Practically everybody is holding high level waste. This is an appropriate time to point out that no country of which I am aware is disposing high level radioactive materials into the ocean. This is generally confused in the public mind. Ocean disposal is being used for certain very limited low level or intermediate level materials, of relatively unimportant radioactive character.

Representative HOLIFIELD. Would it also be true to say that this is not just a national problem, this is an international problem which might well affect the fate of future water supplies for all of humanity, future air contamination and all the other toxic dangers which might be involved unless it is solved on an international basis?

Dr. WOLMAN. That is correct.

Day by day as we expand our own interests in the United States, and as other countries expand theirs, this becomes a problem acutely in need of international understanding, international agreement on criteria and at the moment some form of international supervision, because there will be tremendous competition for location for disposal of materials whether in water, in air, or in the ground.

This cannot be left to any single political entity.

Representative HOLIFIELD. You may proceed.

Dr. WOLMAN. I should like to point out what the situation appears to be, at least up to 1958, in this industry. The industry has somewhat over $25 million invested in the treatment and release facilities for low level liquid wastes. Approximately a billion gallons of low level waste have been discharged per year.

This has been discharged under rigidly controlled, quantitatively monitored basis after, however, in many instances, various forms of treatment ranging from familiar and orthodox treatment and process such as coagulation and settling, to the equally familiar evaporation process.

The total curies so far discharged in low level wastes is over 2½ million.

I need make no comparisons on this as to how that compares with what prior to 1925 or 1930 was the total amount of curies known in the world and in use.

This is a very substantial figure. The only reason it has been released is that it has been released under control and with criteria that would leave the natural environment in a condition which would be without hazard to the general public.

Representative HOLIFIELD. These are confined to short half life materials?

Dr. WOLMAN. These are short half lives, they are the least troublesome.

Even those have to be modified and, are modified, as you will have further evidence during the hearings.

Representative HOLIFIELD. The materials such as cesium 137 or strontium 90, are not released.

Dr. WOLMAN. I would say none of these is released. The solid radioactive wastes such as machine turnings, other contaminated equipment, contaminated trash, laboratory trash and so on, are handled in a variety of ways.

In many instances they are packaged and buried. They are packaged with great care and with strong material casings and are then disposed of at selected points in the Atlantic, the Pacific, and perhaps in the Gulf of Mexico at some subsequent time.

Even in the disposal of these materials of less hazard, where they are disposed of at sea, they have posed and will continue to pose regulatory problems of a governmental nature. They can simply not be given to a collector and left to him to determine where and how he would like to dispose of them.

Present practice is to have this program under detailed supervision of the Atomic Energy Commission, of the Navy and of the Coast Guard at selected points. We stress that control to a considerable degree because our own experience, now running close to 40 years, with disposal of a presumably innocuous material such as garbage into the ocean is not a happy circumstance. We have learned that that is not a good way to dispose even of such materials unless they are very carefully controlled.

You might guess very promptly why that has proven to be the case over the past half century. Scows have taken materials out in storms and they capsize through natural forces. They get upset sometimes through intentional forces so that the haul is reduced. Hence, instead of disposing at a given location of high depth they turn up in redistribution at Atlantic City or Asbury Park or the like.

Those lessons we have learned over many, many years.

As a matter of fact, this particular room was the setting of a hearing on the garbage disposal of New York City at sea in which that entire table was filled with refuse materials from New York City that had their ultimate undisturbed fate on the beaches of Atlantic City and Asbury Park, simply because they did not go to the point of best distribution, if there is such.

This sea disposal is practiced under control and with success, but it does require a high degree of managerial efficiency.

Representative HOLIFIELD. This requires Federal supervision to see that even when these low-level wastes are deposited beyond the Continental Shelf, they are actually deposited at a distance which the regulations require.

Dr. WOLMAN. That is right, at a site which is carefully selected and which cannot be without supervision because all of our history of that kind of disposal, whether garbage or sewage of much more innocuous character, as I need hardly tell anyone from California, cannot proceed without detailed regulations and supervision.

The gaseous radioactive wastes have been superbly controlled by and large.

The investment in devices for removing it are now of an aggregate of some $45 million.

The work in the cleaning of gaseous materials from the atomic energy industry has moved with such rapidity in the development of new ideas, new materials, new fabrications as to have advanced the general art of cleaning of gaseous discharges by at least a half century. This was due to the necessity of rapidly finding ways of handling difficult and dangerous materials.

The treatment and disposal of fuel-processing waste is perhaps the most difficult problem confronting the industry. The industry now stores about 70 million gallons of high-level liquid wastes in tanks.

The investment in that tank storage at the moment is in the neighborhood of $125 million.

Representative HOLIFIELD. Could you give us at that point for layman's understanding the thickness that is necessary in a concrete tank and the type of metal which might be used?

Dr. WOLMAN. They have chosen from time to time almost every type of material. One reason why tank storage in absolute dollars is quite expensive is that they have had to choose material which promised very long life under the exposure of either highly acid or in some instances highly alkaline materials, very high in temperatures, and very high in thermal production of waste itself. The result is that they have moved to metals of high quality and of high cost.

Equipment for those tanks is costly, since the tanks must be provided with cooling facilities for removal of heat.

Representative HOLIFIELD. The nature of the material itself constantly generates thermal heat?

Dr. WOLMAN. Yes. This is again one of the headaches which confronts the industry.

Representative HOLIFIELD. It is not only a matter of putting it in a container and putting it in the ground.

Dr. WOLMAN. You have a problem of heat removal, corrosion, of gradually siphoning off materials that might ultimately be released and the eternal problem of simply holding under those conditions of decay.

These are being held for the decay process and the timing of it is carefully worked out in relation to the constituents of the waste.

It is desirable to give some impression as to where we might be, say in 1980. There are as many predictions as there are people interested in this trade and you will have additional ones during this week.

But on the basis of some of these predictions the accumulated waste volume about 1980 will be of the order of 100 million gallons.

The total fission-product activity in curies by that particular estimate, and I hold no brief for it, will approximate 100 billion which in common parlance is an awful lot of curies.

Mr. RAMEY. Ten to the eighth is 100 million.

Dr. WOLMAN. Yes, 10 to the sixth is the million and so on.

Representative HOLIFIELD. Our investment in these waste handling and disposal facilities today, the Atomic Energy Commission's investment, is around $200 million.

Dr. WOLMAN. It is probably in the neighborhood of $200 million, the best estimate I can get from the Atomic Energy Commission as of

1959. It has been distributed roughly in accordance with the figures which I have already given.

I want to say a word about the prospects for the future with respect to these wastes, not on a quantitative basis, but on a qualitative basis. This prospect for the future is one with which the Commission, its various workers, the private contractors and, of course, the Joint Committee, are primarily interested.

Any evaluation of the prospect of waste disposal in the atomic energy industries must be predicated upon foreseeable technological advances. I read this particular paragraph rather than paraphrasing it because I think it is acute to both the hearings and to the future. These hearings should disclose whether the procedures described are in the category of hypothesis, are in the laboratory stage, with reasonable prospect of being translated into effective use, or are expositions of actual current practice regardless of price.

Since most informed people are concerned about this problem, some tendency is exhibited to announce treatment processes—usually by the newspapers rather than by the technical journals, but they have therefore more damaging results perhaps than they would otherwise have—they have a tendency to announce treatment processes, by product recovery possibilities and forms of disposal which have a high imaginative content, but low ingredients of reality.

Mr. RAMEY. This is like shooting them to the moon?

Dr. WOLMAN. The earlier suggestions, and I think we ought to take credit for that, the earlier suggestion in this country for disposal of these wastes was to shoot them to the moon approximately 15 or 16 years ago. Whatever credit we can take for that I think we ought to take.

There are certain disabilities and certain unresolved problems attached to that proposal, but that is one of the earliest proposals. It has not so far been translated into reality.

The distinction among these various levels of knowledge and of performance should be sharply delineated in these hearings.

I have a feeling, Mr. Chairman, that if the hearings are to serve their maximum purpose they ought to disclose whether these are highly imaginative hopes or whether they are translatable today or tomorrow or even next year into helpful, practical processes.

Some of the technical literature, I might say, has included answers to this which turned out later to have been quite imaginative and nothing more. This might be said with some respect to the problem of recovery, discussed in greater detail in my prepared statement (p. 17). Every industry with which I have dealt over the years has always had, particularly in congressional hearings, the charge that they do not convert their waste materials into usable products. If they did do so, the comment has been that the industry would make all of its money on the usable byproducts and would have to really run the industry in order to make the byproducts.

Relatively few industries of which I am aware show byproduct recovery of that nature either in success or in economy. Where there is recovery it has generally been through the pressure of regulatory agencies; not always with profit, most often with some loss.

I want to add, however, another feature in this instance which again is completely peculiar to the atomic energy industry.

The recovery of, say, cesium 137 and strontium 90 must be distinguished from removal. The recovery processes which take out a considerable part of these two very critical radioisotopes or radionuclides need not have a major impact on disposal or on treatment because in order to have an impact the removal has to be of a very, very high efficiency not yet attained.

Secondly, from the standpoint of administrative officers it has not reduced the problem of administrative supervision; it expands it, because the peculiarities of these products, if recoverable and if, in turn, having a salable value, but more important still, a general value regardless of price, in industry, in medicine, in biology and the like, pose a very large problem of governmental supervision and control.

It poses the problem of inventory, of monitoring of use, and of accounting where these materials are. We are in a dilemma therefore. We are between an upper and lower millstone in these recovery processes, even where they may be successful, that they add to a problem rather than subtract from it.

It may resolve one problem and create another. Obviously, no one would maintain that one should not search for valuable byproducts in these two materials, and others. That search should proceed.

I merely point out to the committee that it will not solve, unless it is extraordinarily efficient, the disposal problem and certainly will not solve the administrative regulatory problem.

I think this is as much as I would want to say, Mr. Chairman. The written document details much of what I have elaborated on and I am now subject to your questions.

Representative HOLIFIELD. Mr. Van Zandt?

Representative VAN ZANDT. No questions.

Representative HOLIFIELD. Mr. Ramey?

Mr. RAMEY. Would you like to comment on Mr. Holifield's opening statement where he mentioned that several of the experts who testified during the fallout hearings stated that the hazards of radioactive waste from the peacetime atomic power industry will be far greater than fallout from weapons tests.

Dr. WOLMAN. I can answer it in two forms. They are related.

From the standpoint of radioactivity inherent in the materials I would say that radioactive wastes have by far a greater total hazard than fallout at this time.

I would say furthermore, however, that since all of these wastes are controlled, are contained, are monitored, are not dispersed, they have from my point of view a very much less hazard than fallout which is dispersed.

Representative HOLIFIELD. Thank you very much, Dr. Wolman. We appreciate your statement. It is well worth reading and we will place the entire statement in the record at this point.

(Dr. Wolman's formal statement follows:)

THE NATURE AND EXTENT OF RADIOACTIVE WASTE DISPOSAL PROBLEMS

(By Abel Wolman, Johns Hopkins University)

In a period of less than 15 years a new industry has developed in the world. A very considerable sector of this industry has grown in the United States to a capital investment now in excess of $6 billion. This atomic energy industry, through the control exercised by the Atomic Energy Commission, has kept the hazards in the disposal of its industrial radioactive wastes well below any appreciable effect upon the industrial worker or the population at large.

As in every other industry which has evolved over the years, the disposal of its wastes has posed hazards and problems. In significant instances many of these problems so far have been unresolved, if we hope for actual disposal (in the familiar meaning) of the materials producing radioactivity.

Wide variation in the nature and characteristics of radioactive wastes is to be considered. Great differences in concentration and kind of radioactive materials are to be found in the gaseous, liquid, or solid state. The volumes to be managed and their physical and chemical characteristics increase the complexity of the problems and the solutions.

The management of radioactive waste materials, under the continuing and careful scrutiny of the Atomic Energy Commission, has followed two general precepts for the protection of man and his environment. The safe application of both of these precepts still remains the central theme of the Atomic Energy Commission policy. These precepts are: with high-level radioactive wastes, concentrate, and contain; with low and intermediate level radioactive wastes, dilute, and disperse to nature.

The utilization of fissionable material in the controlled production of thermal energy or in sudden explosion results in highly radioactive products. The industry produces wastes, at various steps from the mining of the ore to the development of power, with great variations in quality, level of activity, toxicity, and chemical and physical form. The levels of activity range from hundreds of curies per liter in high level wastes to fractions of microcuries per milliliter in low-level wastes.

A characteristic distinguishing these wastes from any other industrial waste, and which makes them perhaps more dangerous as they may be released to the air, the soil or to surface waterways, is that they are not detected by the human senses. With appropriate instrumentation and radiometric analysis, however, the nature and character of these wastes may be determined with accuracy. The toxicity of these wastes to humans is in addition much greater than that of any hitherto familiar industrial poison.

Additional significant differences from more orthodox industrial wastes are the long half lives and the damaging properties to human tissues of certain of the materials produced. The supervision and control of some of these wastes, therefore, must be viewed, not from the standpoint of temporary expediency, but from the necessity of guaranteed supervision and control, in some instances for hundreds of years. Such a contingency places a major responsibility upon and challenge to all the public and private agencies dealing with the problem.

The protection of the public health and of the natural resources of this and of every country, entails a greater depth of continuing responsibility than for any other industrial waste hitherto confronting society.

The total environment of man, therefore, becomes the area of responsible public interest. Since the creation of these wastes is related to the nature of the industrial process, to the environment, to the permanent maintenance of equilibrium conditions for the protection of biological life, every problem of plant location in relation to its environment must be considered. Once the industry comes into operation, the continued and rigid supervision and monitoring cannot be relaxed. Solutions to the problems residing in these wastes and the objectives to be met, in the expanding nuclear-fission industry, demand the participation of the disciplines of health physics, radiation hygiene, sanitary, chemical, electrical, mechanical and civil engineering, biology, chemistry, and physics.

The levels of activity of radioactive wastes from the various industrial processes involved are generally quite low until the fission reaction takes place in nuclear reactors. These hearings will disclose the origin, the nature, and the amount of these wastes from the processing of radioactive ores of uranium to the ultimate preparation of enriched feed material. These processes are chemical and metallurgical in nature, involving in many instances machining of final products. In each instance, wastes result, as the ultimate fuel is produced.

When the fuel is irradiated by neutrons in the reactors, tremendous amounts of heat are produced. Since such reactors are cooled by air, water, liquid chemicals, or other media and since these are in turn exposed to neutrons in the reactors, they become irradiated. These irradiated media must then be ultimately disposed of or managed in a safe manner.

When reactors are disassembled their irradiated parts are removed. A major problem of disposal then ensues. This particular operation requires extraordinary care and the time and amount of exposure of the workers is rigidly controlled. Highly active material is stored in designated areas or is buried. A rigid inventory of such buried radioactive material is maintained. The poten-

tialities of an accident are carefully reviewed and the resulting wastes therefrom are provided for in design and in operation.

The high-level wastes originate principally, but not entirely, from the chemical processing of fuel from nuclear reactors. Since the burnup in most reactors represent a very small percentage of the fuel installed, it is economically essential in present practice to recover unburned portions. Associated with these irradiated fuels which are to be recaptured for further use are the fission products which have resulted from the processes. These have radioactive half-lives varying from seconds to millions of years.

These high-level wastes at the same time contain almost the entire spectrum of fission products when the uranium atoms are split. To make the recovery of fuel and hence the disposal of wastes even more difficult, a variety of salts and acids must be used for the dissolution of the fuel materials. The characteristics of typical reactor fuel processing wastes are therefore complex and pose one of the great problems of the atomic energy industry.

Other problems arise in the production of selected radioisotopes for beneficient uses in medicine, in biological laboratories, in industry, and in general research. These uses have grown tremendously in the last 10 years. As they find their way into almost every form of activity, problems of supervision, inventory, and disposal arise. The transport of these materials across the country, the continuing control over their distribution, the insistence upon adequate means of disposal for the protection of the worker and the public have created a vast governmental obligation. The exercise of this obligation has been surprisingly good, even though minor lapses have occurred which merely point to the necessity for continuing intensity of supervision. This degree of governmental responsibility is unlikely to decrease. Although the levels of activity with which the country is concerned in radioisotope use is relatively low in comparison with those already commented upon, the radioisotope for universal application must not be ignored in public responsibility.

PRESENT QUANTITIES OF WASTES

In 1958, the industry has somewhat over $23 million invested in treatment and release facilities for low-level-liquid wastes. Approximately 1.5^9 gallons of low-level wastes have been discharged during the year on a rigidly controlled, quantitatively monitored basis, after various forms of treatment ranging from orthodox waste-treatment processes, such as coagulation and settling, to evaporation. The total curies discharged in low-level wastes to date is somewhat over 2,600,000 curies. A summary of the low-level-liquid waste situation is shown in table 1 as of the year 1957.

For the most part, solid radioactive wastes, such as machine turnings, contaminated equipment, and contaminated trash have not constituted a serious technical problem. They have either been buried in the ground under controlled conditions or have been disposed of at sea. The materials so disposed of at sea have been small in radioactive contribution. The problem now confronting the industry with respect to solid wastes is the location of additional centralized handling, disposal, and burial areas. Costs are not excessive, but they are high.

Gaseous radioactive wastes result from many operations, as air-cooled reactors, chemical-processing plants, and fissionable-material-fabrication facilities. In these instances the levels of activity and the concentrations of pollutants vary widely with the type of operation. Facilities for the treatment and handling of gaseous wastes so far represent a total investment of some $35 million.

The treatment and disposal of fuel processing wastes represent perhaps the most difficult, expensive, and so far unsatisfactory features of the entire problem. Up to the present time, approximately 65×10^6 gallons of high-level-liquid wastes have been placed in tank storage. The investment in tank storage now represents something over $115 million. A brief summary of the high-level-waste situation is given in table 2.

Various estimates have been made of high-level wastes which will result from future nuclear-power generation, based on estimates of the extent of such future nuclear-power production, fuel-irradiation levels, and unit volumes of waste per unit of fuel processed. Without attempting the role of a prophet, and on the assumption that nuclear power-generating capacity may attain in 1980 something of the order of 1.1×10^5 megawatts of heat, the accumulated wastes volume in gallons will be of the order of 10^8. The total fission-product activity in curies will then be approximately 10^{11}. These assumptions are shown

in table 3. Supplementary data of pertinent interest are shown in table 4. The future situation will be discussed specifically in the course of these hearings.

All of these figures simply demonstrate that the industry will be confronted increasingly with a waste disposal and management problem, which one is unlikely to escape in the present state of the art. Although a great deal of attention is now being paid to this problem, to many industrialists the problem of waste disposal appears to be nonexistent. The reason for this happy state of mind lies in the fact that under present procedures the Atomic Energy Commission holds itself responsible for the handling of these most difficult materials.

The total investment in waste handling and disposal facilities within atomic energy operations already approaches $200 million, of which almost two-thirds represents capital investment in tanks and appurtenances for long-term retention of high-level-radioactive wastes.

PROSPECTS FOR THE FUTURE

Any evaluation of the prospect in waste disposal in the atomic energy industry must be predicated upon foreseeable technological advances. The hearings of this joint congressional committee should disclose, however, whether procedures described are in the category of hypotheses, are in the laboratory stage, with reasonable prospect of being translated into effective use, or are expositions of actual current practice. Since most informed people are concerned about this problem, some tendency is exhibited to announce treatment processes, byproducts recovery possibilities and forms of disposal, which have high imaginative content but low ingredients of reality. The distinction among these should be sharply delineated in these hearings.

A large amount of effort is being expended in research and development toward waste treatment. Very little, however, is yet available in large-scale practical ultimate disposal systems. The findings to be presented may lead to more optimistic conclusions. At this writing, however, the safe dispersal of large quantities of radioactivity in the environment is not only remote but improbable. It is undoubtedly the part of wisdom to contain and to control the more hazardous and perhaps the potentially useful materials.

The technical feasibility of direct disposal of high-level wastes into specific geologic formations is sufficiently clear to drive one toward at least limited application of such a procedure, either in salt domes or salt beds, in deep, isolated basins, in shale formations, or even in porous formations. It would certainly be inadvisable to move to these presumed solutions to the problem, without far more extensive exploration and pilot-plant application than are yet at hand. The engineering and economic issues involved are still in the very preliminary stages.

The artificial fixation of radioactive materials into stable, solid media has now been experimentally studied for some 4 to 5 years. The objectives in each instance are to fix materials of high hazard in media of nonsoluble, nonleachable character, so that these materials might then be placed or stored in manageable size and character so that their hazard to surrounding environments would be almost nil. Here, too, these efforts may not yet be adjudged to have demonstrated either engineering or economic feasibility. The findings to be presented at these hearings may modify these conclusions in a more favorable direction.

The recovery of valuable byproduct materials, such as particularly cesium 137 and strontium 90, would have two important assets. The selective high efficiency removal of such specific nuclides from high-level wastes would reduce somewhat the high-level waste disposal problem. It would not eliminate it, but it would make the holding of such materials less difficult in time and in risk. Such separation might produce materials of value to society. Unless the separation or recovery, however, was very high, the waste disposal problem would not be much relieved. One must distinguish between *recovery* and *removal;* for example, 95 percent Cs^{137} recovery might be fine for fission product utilization but does not do much good for waste disposal directly. The search for valuable byproducts of industrial wastes is interminable in this and in other industries. Historically, it is unfortunately true that promises of great financial returns from waste recovery byproducts or solutions to waste problems through byproduct use have fallen far short of fulfillment in most industries.

Even if these hopes of byproduct recovery and use should be fulfilled in the case of radioactive materials, the problem of management and of governmental supervision not only will not be reduced by such procedures, but in fact may

be greatly increased because of the wider use and dissemination of highly toxic materials.

A general recapitulation of the directions of effort in the treatment of high-level wastes is given in table 5.

The disposal of high-level wastes in the ocean has had much discussion. It is not unfair to conclude in the present state of understanding of the ocean that such a procedure is unlikely to be sanctioned for some years to come. So many unknowns and unpredictables with respect to oceanic behavior still remain so that management and Government will be driven to the safer practice of keeping high-level radioactive materials under more obvious scrutiny and control than would be provided by most ocean disposal procedures.

Sufficient has already been said in this presentation to make clear that the rapid development of the atomic energy industry is in no small measure contingent upon more prompt and more complete answers to the waste disposal problem. If the situation is somewhat pessimistic, it is only because sufficient energies have not yet been expended in developing economic and safe improvements in waste handling and disposal processes. Progress in this field is unlikely to occur, if the problems are not realistically confronted. It is a tribute to the Atomic Energy Commission operations that so many ad hoc solutions have been provided which have safeguarded the public during the last 10 to 15 years. It is, however, equally clear that approaches to the management and disposal of these materials will require increased attention, expenditure of large amounts of money for research and development and the integration of multidisciplines for accomplishing a more satisfactory long-term answer than is now at hand.

Simultaneously with such emphasis, increased attention will necessarily be devoted to the location and type of reactors and chemical reprocessing facilities, the location of suitable ultimate disposal sites, the shipment of highly radioactive materials and associated hazards, and equating economic validities with all of the less tangible, but equally important, consideration herein discussed.

Much thought will be required in the foreseeable immediate future as to the best way of continuing long-term responsibility for disposal of radioactive materials. It is not inevitable that in this responsibility Government will always have to play a dominant role, except in a supervisory capacity. Industry must increasingly assume responsibility for the physical operation of waste disposal plants under the criteria to be established promptly by Government. The selection of sites for nuclear energy facilities is closely related to waste handling and disposal operations. The recognition of this close relationship is not yet dramatically obvious to many individuals concerned with the development of this industry.

No student of this field can escape the additional conclusion that waste handling and disposal has international as well as national aspects. Technological solutions made in this and in other countries will make a major contribution to these international answers. This is peculiarly an area where political boundaries have little or no relation to the waste disposal problem. This is particularly true in the dispersal of materials to the environment via the atmosphere and the surface waters. From both the national and international standpoint, one cannot ignore the increasing problem which will result from the rapidly extending use of mobile reactors, such as submarines, surface ships, and aircraft. The problems resulting from these operations are rapidly appearing on the horizon and are not specifically developed in oral presentations although they will be noted to some extent in the record of the hearings. This does not mean that they will not emerge rapidly in the not too distance future and should not be overlooked.

TABLE 1.—*Low and intermediate level liquid waste data*

(*a*) Investment ($1,000)	45,000
(*b*) Volume discharged per year (excluding 35×10^6 HAPO) (10^6 gallons)	1,530
(*c*) Total curies discharged (10^3) (to date)[1]	2,600
(*d*) Annual operating and monitoring costs ($1,000)	2,000

[1] Approximately 95 percent of total activity (*c*) is associated with Hanford operations.

TABLE 2.—*High level liquid waste data*

(*a*) Wastes in storage (10^6 gallons)	65
(*b*) Total tank capacity (10^6 gallons)	110
(*c*) Cost—tanks and appurtenances (10^6)	115

TABLE 3.—*Estimates of future wastes (summarized from Wash-742, Lieberman, August 1957)*

	1965	1980
(*a*) Nuclear generating capacity (MW heat)	2×10^4	1.1×10^5
(*b*) Fission products (curies)	2×10^{10}	10^{11}
(*c*) Processing capacity (tons per day)	5	25–80
(*d*) Accumulated waste (gallons)	5×10^6	$1–3\times10^7$

NOTE.—Based on irradiation of 4,000 MWD per ton U and 1,000 gallons high activity waste per ton U.

TABLE 4.—*Example of status of shipment of irradiated fuel and high-level wastes in 1980 (After Wash-742, Lieberman, August 1957)*

(*a*) Megawatts of heat-stationary reactors	1.1×10^5
(*b*) Fission products, curies per ton U:	
After 100 days cooling	5.2×10^4
After 2,000 days cooling	8.0×10^3
(*c*) Megawatt days per ton, burn up	4,000
(*d*) Tons U per day processed	27
(*e*) Gallons liquid waste per ton U processed	1,200
(*g*) Fission products, curies in transit (fuel)	189
(*h*) Gallons of waste in transit	9.8×10^3
(*i*) Fission products, curies in transit (wastes)	227,000
(*j*) Fission products, curies per gallon of waste	660

NOTE.—Fuel shipped after 100 days cooling. Wastes shipped after 2,000 days cooling.

TABLE 5.—*Possible methods of treatment of fuel reprocessing wastes prior to ultimate disposal on land*

(*a*) Decay storage
(*b*) Precipitation
(*c*) Neutralization
(*d*) Evaporation
(*e*) Ion exchange
(*f*) Solvent extraction
(*g*) Crystallization
(*h*) Solid fixation
(*i*) Recovery of byproducts

Representative HOLIFIELD. Our next witness is Mr. Floyd L. Culler, of Oak Ridge National Laboratory, who will speak to us on the nature of waste—source, quantity, and characteristics.

Mr. Culler, we are happy to have you here this morning. You may proceed with your statement.

STATEMENT OF FLOYD L. CULLER,[2] OAK RIDGE NATIONAL LABORATORY

Mr. CULLER. Mr. Chairman and members of the committee, I am indebted for information to a number of people who have prepared papers for the record; I will attempt to summarize data contained in these papers and to bring out the characteristics of the radioactive waste materials.

Radioactive wastes, as Dr. Wolman has pointed out, are generated at each step in the nuclear-fuel cycle. I shall attempt to describe the

[2] Education: B.S., chemical engineering, 1943, the Johns Hopkins University.
Professional history: Associated with atomic energy since 1944. From 1947 to present: Oak Ridge National Laboratory operated by Union Carbide Nuclear Co. Present position: Director of Chemical Technology Division. In charge of research and development in the nuclear energy field, covering four major types of effort: (1) chemical development and research, (2) unit operations development, (3) pilot plant programs, and (4) design and economic studies.
Related activities and experience: National Academy of Sciences, a member of subcommittee to study the disposal and dispersal of radioactive wastes (1955 to present). Processing Advisory Committee to Idaho Operations Office, USAEC (1949–53). Program Committee for Chemistry and Chemical Technology Sessions for the International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland (1955). Participated

types of materials that are generated at each stage in the fuel cycle and to give estimates of the existing quantities.

I shall not make predictions concerning future wastes; these are planned for later discussion by another speaker.

There are three principal sources of radioactive materials; the first being those which occur in nature. From the standpoint of the nuclear-energy business, for the production of power with which all of us are concerned, the most significant of these are the naturally occuring isotopes: (1) U^{238} and its decay daughters; (2) U^{235} which is present in natural uranium and the only naturally occuring fissionable material. It occurs to seven-tenths of 1 percent. Its decay daughters are not significant contributors to the activity of natural uranium.

(3) Thorium 232 is the other naturally occuring radioactive material. Thorium 232 as an isotope is 100 percent abundant in nature. It decays with a very long half life of 1.39×10^{10} years. It is slightly more hazardous than the natural uranium, more care must be exercised in handling it.

The second general source of radioactive materials occurs from the fission process itself. The principal products here are the fission products of U^{235}, of U^{233}, or of plutonium 239.

Describing the characteristics of the fission products would take many hours. I have chosen several out of the some 37 fission-product elements as significant in waste disposal. These are listed in my prepared statement along with an estimate of the relative activity after various periods of decay, following discharge from the reactor. Cesium 137 has a 26.6 year half life. As a result of this it is one of the most significant long-term fission products and will remain as a principal radioactive isotope in waste for a long period of time.

Strontium 90, with a 28-year half life, is a principal contributor to the long-term activity of the waste and is the most serious biological hazard in the fission product spectrum.

The rest, including cerium 144, krypton 85, iodine 131, zirconium, barium, ruthenium, have shorter half lives and appear in long-cooled wastes in much lesser quantities than cesium and strontium.

The second class of materials that are produced by fission in a reactor result from the parasitic capture of neutrons by the fissionable materials themselves, or by the fertile isotopes, U^{238}, and thorium 232; or by structural materials in the reactor; or by the coolant and coolant impurities.

Neutrons from the fission of plutonium 239, U^{235} and U^{233} can be captured to produce other heavy elements of the same series, such as U^{234}; plutonium 240; and plutonium 241; plutonium 238; U^{232}; U^{237}; americium; and curium.

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in same, attended and presented paper. United States delegate to the French Atomic Energy Commission and to the United Kingdom Atomic Energy Authority (August 1955).

Memberships (professional): American Nuclear Society, American Institute of Chemical Engineers, American Chemical Society.

Special assignments: Belgian Symposium on Chemical Processing in Atomic Energy, Brussels, 1957. Program chairman, author of papers, attended. Currently, chairman of the Division of Industrial and Engineering Chemistry of the American Chemical Society. Currently, ANS representative to the N–5 committee of the American Standards Association in the chemical engineering field of the nuclear energy field. Participated on the Chemical Engineering Review Committee and was author of paper for the 1958 International Conference on the Peaceful Uses of Atomic Energy, Geneva (1958).

Personal: Married; one son. At present engaged as chairman of the Oak Ridge Regional Planning and Zoning Commission. This is the first group to function officially as a group of private citizens planning for the future of Oak Ridge as non-Government-supported community.

These heavy elements, because of their long half lives and their high biological hazard, are significant contributors to the hazards of wastes.

To illustrate how neutrons are consumed in the production in a reactor and what percentages go to fission products and heavy elements, the neutron balance sheet in my prepared statement is offered.

Each gram of uranium or plutonium fissioned releases about 201 Mev. of energy.

This amounts to 24,000 kilowatt-hours if the gram is fissioned in 1 day.

The major portion of this energy, about 80 percent, appears as heat generated by fast moving fission fragments. Six percent of the energy goes to the fission products, which decay over varying periods of time.

To illustrate the rapidity with which the energy in fission products decays, consider the case in which uranium is irradiated in a reactor for 350 days at 3×10^{13}n/cm^2/sec; from this irradiation the total fission product power is approximately 38½ watts at the time of discharge from the reactor; this energy release is reduced by a factor of 10 in 3 hours to a 3.85 watts per gram of U^{235}; a factor of 100 in 58 days; a factor of 1,000 in 520 days, and a factor of 10,000 in 4,000 days.

After approximately 4,000 days, the energy being emitted in the collected high-level waste is due almost entirely to cesium and strontium, the principal long-lived-fission products.

In a reactor there are several processes for utilizing neutrons. The neutrons produced during the fission process interact and are ultimately captured by the atoms which surround the fissionable material.

There are several other processes which could occur, in fact, there are four possible processes for interaction:

(1) the complete loss of neutrons from a nuclear reactor. This loss, of course, must be kept to a very minimum for efficient reactor design.

(2) fertile and parasitic capture; neutrons which produce new fissionable materials or new heavy elements are captured by the fertile and fissionable nucleii.

(3) nonfission capture, by structural materials, coolants and contaminants.

(4) and, of course, capture leading to fission.

If one starts with 100 neutrons in a reactor, approximately 5 leak out during the period of time in which they are slowed down. Ten are captured in U^{238}. Eighty-five of them are slowed down to the point where they can effectively enter into low energy neutron reactions. Five of these neutrons are available for absorption parasitically by the structural materials, 33 are absorbed by U^{238}, other heavy elements, moderator, poisons, and 7 by the nonfission capture by U^{235}.

Representative HOLIFIELD. Mr. Culler, the Chair will tell you that you are going into this thing in too much technical detail. Your complete statement will be accepted in the record and it is valuable, but I think in its scientific impact, this is beyond the comprehension of most of the members of the committee.

So could you go from the detailed scientific part of it into the description of wastes and the other problems which will be more understandable to us?

Mr. CULLER. I am sorry, Mr. Chairman. If you will turn to my prepared paper, the first step in the fuel cycle chain starts with the mining of fissionable and fertile material.

Uranium is recovered as a product from many mines and 26 mills in the western part of the United States, plus a very small operation in the Florida leach zone. The western mills range in size from 50 to 3,500 tons, and they produce roughly 21,000 tons of ore per day.

The waste from the uranium mines themselves are not significant; the problem principally involves the protection of the miners from radioactive dust and radon gas.

Harmful concentrations of airborne radioactive materials can be removed from the mines by proper ventilation. The air exhausted from the mines does not present a problem of general atmospheric contamination.

At the uranium mills, however, there are wastes produced by the milling operation for the recovery of uranium which contain fractions of the uranium content and the decay daughters of uranium itself. Disposal of mill wastes presents severe local problems since large volumes of water are produced—approximately 800 gallons of waste per ton of ore treated, or approximately 18 million gallons of water per day.

Check analyses have shown that the natural uranium content in these wastes which are now impounded and slowly bled to the surrounding streams, is below the maximum permissible tolerance.

However, a problem does exist with concentrations of radium 226 which is a decay daughter of uranium.

Methods of treatment are being pursued. I think that the problem of disposal of waste from the mills will be satisfactorily solved.

The next step in the feed cycle after the uranium milling operation, from which uranium is produced as a precipitated oxide, is additional purification in the feed materials step. Feed material preparations are performed by Mallinckrodt of St. Louis, and Weldon Springs; by the National Lead Co. at Fernald; at the Oak Ridge gaseous diffusion plant to some extent; and a new operation by the General Chemical Co.

In this stage of operations the problems associated with waste are quite similar to those at the mill, since only natural uranium and possibly thorium are handled. The radioactive wastes are the decay daughters of uranium. The volumes of water are large and the waste problem is not a serious one, but it requires the control of small quantities of radioactive materials in large quantities of water.

This problem may not always remain the same as the nuclear power economy develops because of the built-up isotopes of uranium and other heavy elements which will follow through the recycle to appear as more hazardous trace elements in the plant wastes.

I will now describe briefly the radioactive wastes produced from operating reactors other than fuel and fission products. The nonfission capture of neutrons by coolants, impurities in coolants, by structural materials, and by corrosion products produces radioisotopes which must ultimately be discharged from the reactor or decay. Also, fission products can be released from ruptured fuel elements under unusual conditions. Usually, however, the fission products and new heavy elements produced by fission and by fertile or heavy element absorption of neutrons, remain within the fuel elements until they are dissolved in the radiochemical plant.

In water-cooled reactors the radioactive nuclides produced by the irradiation of water itself, neglecting, for the moment, impurities, and corrosion products, produce radioactive nuclei of very short half life, but of sufficient radioactive intensity to control the thickness of the shielding around the circulating water loops of the reactor. The principal isotopes from water alone are nitrogen 16 and 17. These decay with very short half lives and can be controlled by retaining the cooling water for a short time.

Long-lived radioactive isotopes in water cooling systems result primarily from the activation of chemical components of structural materials or fuel elements in the core; they appear as soluble form or as precipitates in the coolant. Some of the irradiation products are deposited in the external cooling system of the reactor, thus increasing the penetrating radiation from these systems.

The products causing the greatest difficulty here are cobalt 60, which results from an impurity in stainless steel, and Fe–55, because of their long half lives.

Here again the problem of waste disposal is one of handling large volumes of waste water with almost insignificant quantities of radioactive materials. Wastes from water-cooled reactors cannot be ignored since they will present local problems of control.

There are other reactor coolants which are now being used or considered. One is liquid sodium. The sodium coolant becomes highly radioactive, as a result of irradiation with neutrons. However, radioactive sodium has a short half life and can be allowed to decay within the cooling circuit itself.

Irradiated reactor materials are chemically processed to reclaim the unburned nuclear fuel and recover the transmutation products, such as plutonium 239 and U^{233} from mixtures of the fission products and from the inert components of the fuel. At the present time these materials are separated by a process called solvent extraction which produces wastes which are highly acidic and in some cases have large concentrations of nonradioactive components. The wastes are almost always aqueous solutions or dilute suspensions of 99.9 percent of the fission products.

In the solvent extraction processes, solid fuels, which have been stored 90 to 120 days following removal from the reactor to permit decay of short lived fission product activities, are dissolved in nitric acid and the solution is fed to a solvent extraction column. The wastes containing more than 99 percent of the dissolved fission product and inerts in the fuel and scrub solutions leave at the bottom of the column.

The uranium and plutonium are selectively extracted into an organic liquid, then separated from each other and decontaminated further from the fission products by additional treatment.

There are several solvent extraction processes now in use. Principal among these are Purex, which produces approximately 990 gallons of high-level fission product waste per ton of uranium processed. This volume can be reduced by evaporation to approximately 80 gallons of highly concentrated waste per ton.

A second process which has been used is Redox, which produces approximately a thousand gallons of high-level liquid waste per metric ton of uranium and contains a dissolved salting agent, aluminum nitrate, which prevents its volume reduction to a very great extent.

Several other processes are possible and are being used such as the tributylphosphate-U^{235} process, and the Thorex process, for removal of U^{233} from irradiated thorium.

In table 1, page 45, the characteristics of the first cycle high-level aqueous wastes are given. It should be emphasized that these numbers are all approximations and that the actual values depend on the operating characteristics of the particular chemical plant and the details of the chemical flow sheet being used.

In addition to the concentration of the inert, nonradioactive chemicals given, all the wastes contain fission products and to a lesser extent, the uranium, plutonium, and other heavy elements that are lost in the process.

The fission products and heavy elements in wastes from the Purex process add little to the total solids content of the waste; waste properties are essentially those of nitric acid. In addition to the fission products there are trace constituents of iron, nickel, and chromium present in various amounts. Since Purex wastes have few dissolved solids they can be reduced greatly in volume.

The waste from most other solvent extraction in processes contain significant quantities of dissolved salts of aluminum. In the Redox process, this aluminum is added to assist in the recovery of uranium and plutonium. In the processes for enriched uranium it is part of the original constituent in the fuel.

In the Thorex process it is added to assist in fission product decontamination.

The total stored volume of high level wastes of the nature described above is approximately 72 million gallons; available tank storage capacity is approximately 103 million gallons.

Although the fission products are the source of most of the radiation hazard they do not greatly influence the chemical properties of the wastes because they are present in such low concentrations. For every megawatt-day of irradiation, about 1.1 grams of fission products, both radioactive and stable, is formed. On this basis it can be seen that most of the treated wastes contain on the order of one gram per liter of total fission products consisting of the isotopes of some 37 of the elements.

While the total number of the fission product atoms does not change after irradiation, the concentration of specific fission products does change with time, and for any given fuel is a function of the reactor power level during irradiation, the length of time that the fuel has been irradiated, and the time elapsed since the discharge from the reactor.

The contribution to the total heat-emitting activity of fission products of 10 different mass numbers is given in figure 2, on page 48; here again it can be seen that at a decay time of 10 days, the 63-day zirconium 95-niobium 95 account for 20 percent of the total beta activity. They are followed in decreasing order of significance by barium 140-lanthanum 140; 290-day cerium 144 and its daughter, 17½ minute praseodymium 144.

One of the properties of the fission product mixture of most importance in disposal, is their heat emission. In any disposal scheme provision must be made to dissipate the heat evolved as a consequence of radioactive decay if elevated temperatures, which can lead to chemical instability, corrosion, and entrainment during storage are to be

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avoided. If it is assumed that 800 gallons of wastes are produced from each metric ton of uranium processed and from this ton of uranium 10,000 megawatt-days of heat have been extracted, the fission product heat production is somewhat formidable. After 1 day's decay 1 gallon of waste will be emitting 1,700 watts. But after 100 days' decay, a reasonable time to allow before processing, this power will have dropped to 200 watts per gallon.

The power emission continues to fall until a decay time, say, of approximately 3,300 days (or 9 years) when the rare earths have begun to drop to a level where strontium and cesium control the power.

At this point the total power emitted by fission products amounts to 0.5 watts per 1 gallon and decreases according to the 27 to 28 year half life of cesium 137 and strontium 90.

There are other processes that are being studied as possibilities for recovering the fissionable and fertile materials from the fission products. At the present time and probably for some time to come, however, all radiochemical fuel reprocessing, which generates the high-level wastes, is being accomplished by organic solvent extraction. The existing solvents can and will be used to recover fissionable and fertile materials from feed solutions prepared by the variety of techniques required for power reactor fuels. The solvent extraction processes are likely to be the principal means of radiochemical separation for some time to come.

It is probable also that large central reprocessing chemical plants will process the fuel from many power reactors, probably 10 or more. The economy size of a radiochemical plant may be of the order of approximately 6 tons per day capacity for natural uranium or equivalent.

In addition to the organic solvent-extraction methods, new processes are being developed from which the high-level wastes will be slightly different than from solvent extraction. For reactors of high fuel and fertile material content such as the fast breeder where the inventory charges on the fuel are high, processes which give partial fission product decontamination are being studied. The pyrometallurgical processes being studied at Argonne for EBR II are examples of this type of process. Wastes from the fuel melting and slagging or from salt washing will be produced principally in solid form. However, it is difficult to define the exact nature of these wastes at the present time because of this state of the development of the technology.

Another process which has been operated successfully on a large pilot-plant scale at Oak Ridge and Argonne is the fluoride-volatility process. Here the waste will be produced as a component of a fused fluoride salt mixture. Again wastes are produced in solid form.

So far I have described primarily the high-level-liquid wastes which emanate from a radiochemical separations operation. In addition to these, there are active gases that are produced at the radiochemical separations plants. The fission-product gases which remain after a normal cooling period of a hundred days are krypton 85, which is a noble gas with a 10-year half life, iodine 131, in very small concentrations, with a half life of 8 days.

If the processing of reactor-fuel elements is started before a cooling period of 90 days, another gas, xenon 133, will appear in the waste-gas stream.

Iodine must be removed completely from the off-gas streams; processes now exist that are 99.9 percent efficient for iodine removal. Iodine is removed from the waste gases by passing it over a bed of silver nitrate coated ceramic saddles at elevated temperatures or by scrubbing with a caustic solution.

Dissolver gases contain, along with fission products and air, gaseous oxides of nitrogen, and possibly other acid gases.

The table on page 52 gives a rough estimate of fission gas quantities per gram of U^{235} consumed. If natural uranium is irradiated in an average flux of 3×10^{13} n/cm^3/sec for 1 year, for each gram of U^{235} destroyed, approximately 0.3 curie of krypton 85 will be present after 20 days' decay; and approximately the same quantity after 100 days of decay. However, the same is not the case for the shorter lived iodine. If there are 10 curies after 20 days, only a hundredth of a curie remains after 100 days. For xenon, 11 curies exist after 20 days; and a thousandth of a curie after a hundred days' decay.

In addition to the radioactive fission product gases, small particles which contain radioactive elements are entrained and present a very serious problem of gas cleaning. The radiochemical plant is a major producer of these aerosols, many of which have high-specific activity. Many operations in the chemical process produce mists or entrain solids. These can be effectively removed, however, by filtration and other treatment techniques.

Representative HOLIFIELD. Mr. Culler, can you give us a qualitative recapitulation of the relative degree of difficulty of the waste-disposal problems in three fields, feed material, reactor operation, and the chemical process?

Dr. CULLER. The feed-materials operations is characterized by very large volumes of very low-activity waste. The problem of their disposal is not a difficult one, from a long-term point of view, because very little radioactive material is contained in the wastes. However, disposal of the slightly contaminated large volume wastes must be carefully monitored.

The discharge of radioactive gases and liquids from reactors will not present a serious long-term waste disposal problem. Reactors can be operated so as to prevent the release of long-lived radioactive materials to the environment, barring a reactor accident.

The major source of radioactive materials will be the radiochemical plants where the high-level-liquid wastes are produced and from which fission gases and particulates are generated.

Representative HOLIFIELD. That is where the real danger lies in the processing of these fissionable fuel elements?

Dr. CULLER. Yes, sir; and the wastes which result from these.

Representative HOLIFIELD. Where are the locations of the processing plants?

Dr. CULLER. The principal processing plants are at Hanford, Savannah River, Idaho Falls, at the National Reactor Testing Station, and minor operations at Oak Ridge National Laboratory and Argonne.

Representative HOLIFIELD. Will you comment on the radioactivity associated with gas-cooled reactors and with air-cooled auxiliaries of the other types?

Dr. CULLER. In most reactors the radioactivity associated with the circulating coolant is contained in the circulating system for decay;

usually it is not much of a disposal problem because of the short-lived nature of the induced radioactivity.

If gas coolants must be vented, the radioactive-material content can be reduced by decay storage, filtration, or chemical treatment.

Probably gas-cooling systems have a higher potential for releasing activity in the event of a reactor accident, but I think this type of incident and its results are somewhat out of the scope of his hearing.

Representative HOLIFIELD. At this time the Atomic Energy Division of Inspection is responsible for maintaining a good practice in the handling and disposal of radioactive sources.

In your experience do you believe they have an adequate staff to do this job properly?

Mr. CULLER. From my personal experience, I think they are doing an adequate job. They have recently pointed out the problems of disposing of slightly radioactive mill wastes on the Colorado Plateau and are working effectively with the Atomic Energy Commission Division of Raw Materials in solving this problem.

Representative HOLIFIELD. In the transportation of spent materials from the reactors to the chemical-processing plants, are you of your own knowledge able to say that adequate precautions are taken against traffic accidents and other types of hazards which might occur in shipment?

Mr. CULLER. In answer to that question it might be best to observe that no serious accidents have occurred in the transfer of materials between one site to another. We have shipped both canned-fuel elements in adequately shielded containers and liquid wastes from Hanford to Oak Ridge and from Argonne to Oak Ridge and other sites. No accidents have occurred.

However, the evaluation of the statistical probability of an accident and the evaluation of its results is a difficult one. I think that adequate precautions are taken at present in the transfer operation to prevent serious spread of contamination; the problem will change, however, as the industry grows.

Representative HOLIFIELD. I suppose that shipments are kept to the absolute minimum, there is no wholesale transfer of materials from Hanford to Oak Ridge from the standpoint of storage? It is only from the standpoint of research and development, and that sort of thing; is it not?

Mr. CULLER. Primarily that.

However, in a truly economic nuclear-energy business, because of inverse dependence of chemical-reprocessing costs on plant size, it will undoubtedly be necessary to ship radioactive-fuel elements from several reactors to a central chemical-processing plant.

Representative HOLIFIELD. When that occurs, it may be necessary to have more storage plants and more locations in the United States?

Mr. CULLER. The number of plants that will exist will be determined by the allowable cost for the chemical processing operation. These costs are greatly affected by the chemical plant size. Since the number of reactors that must supply a given baseload for a chemical plant is large, it is probable that there will be a few chemical plants and many reactors.

For instance, we have estimated that there will be possibly 20 chemical plants and many hundreds of reactors. The chemical plants

can be centrally located so that the shipping distances are not greater than 200 to 500 miles. Their location must take into account the problems of waste disposal and shipping difficulties.

However, there will be a problem of cooling and containing the transported fuel elements at a time when a nuclear economy does develop.

Representative HOLIFIELD. Do you think that the increased use of radioactive sources and tracers will offer a communitywide hazard or do you think it will be confined to the people who use them in industrial plants?

Mr. CULLER. Depending on the nature of the radioactive elements used, it can vary, but certainly can be a communitywide problem.

Representative HOLIFIELD. We had an illustration of that at Houston a year or so ago where improper use in an industrial plant not only created a hazard in the plant itself, but it was carried into the community.

Mr. CULLER. Yes.

Representative HOLIFIELD. Have the rules and regulations along that line been tightened up since that time to the point where this is less likely to happen?

Mr. CULLER. I cannot answer that from my own personal knowledge. My impression is that the rules and regulations were adequate to offer protection in that event but that proper controls were not used where the acccident occurred; that is, rules and regulations were not complied with, to some extent.

Mr. TOLL. Mr. Culler, is it correct that next to any chemical processing plant there will have to be nearby storage facilities for these very high level wastes?

Mr. CULLER. Yes, that is correct.

The chemical plant and at least some high level storage facilities will almost surely have to be built together.

Mr. TOLL. The location of these chemical processing plants would be one of the most important factors to take into consideration then, in the future?

Mr. CULLER. It is an important factor. The chemical waste storage tanks will serve as accumulators for all of the fission products and heavy elements generated in all of the power reactors, and as such will store up, over a period of time, greater equilibrium quantities of fission products and heavy elements than will the reactors.

Mr. TOLL. In your opinion, Mr. Culler, has sufficient consideration been given in the location of the present plants to all necessary factors and hazards involved?

Mr. CULLER. Yes.

This is obvious, I think, from the fact that no serious releases of radioactive material affecting the general population have occurred. From having been associated with chemical plants for some time, I know that great care is taken in selecting the sites for remoteness and for proper geological conditions.

Representative HOLIFIELD. Did you have anything else to add?

Mr. CULLER. I think that is all, Mr. Chairman.

Representative HOLIFIELD. Thank you very much, sir, for your statement. You have a very complete statement here which will become a very valuable adjunct to our hearings.

(The formal statement of Mr. Culler follows:)

NATURE OF RADIOACTIVE WASTES

by
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Radioactive materials in waste streams from the many processing steps in the nuclear energy industry present a new problem in environmental pollution control. New techniques and standards for the handling and disposing of liquid, solid, and gaseous wastes have been developed within the past fifteen years to meet the difficult control problems presented by these radioactive wastes. During this period, technological foundations have been established for a nuclear power industry which will surely develop into a major contributor to the energy production of the United States and the world.

Although the exact nature of the reactors and fuels to produce economic nuclear power has yet to be determined, the fuel cycle and the resulting radioactive wastes are predictable, based on our present knowledge.

The many integrated processes of a nuclear industry must be considered to define the nature of wastes that are being and will be produced.

All of the processes for the recovery and preparation of uranium and thorium as fuel produce wastes containing radioactive materials in more concentrated form than exists in nature so that problems of confinement or dispersal are introduced. New radioactive species are produced by the fission process, or by the capture of neutrons which result from fission by elements in a fission reactor.

The radioactive materials that must be considered are:

(1) Naturally occurring feed materials.

(2) Fission products.

(3) Radioactive elements produced from parasitic neutron capture.

Naturally Occurring Feed Materials

(1) U^{238} and its natural decay chain. 99.28% of natural uranium, decaying with a half-life of 4.51×10^9 years. In nature it can be considered to be in equilibrium with its decay daughters.(3) Important isotopes to be considered in wastes are U^{238}; Th^{234}(24.5 days)-Pa^{234}(6.7 hrs); Th^{230}(8.3×10^4 yrs); Ra^{226}(1590 yrs); Rd (3.8 days); possibly Po^{210}(140 days).

(2) U^{235} and its natural decay chain. 0.71% of natural uranium, decaying with a half-life of 7.07×10^8 years. Not a significant contributor of hazardous radioactive species because of low concentration.

(3) U^{234} and its decay daughters. 0.005% of natural uranium. Decay products are not significant because of low concentration in natural uranium.

(4) Th^{232} and its decay chain. 100% abundant in nature. Decays with a half-life of 1.39×10^{10} years. Important decay daughters are Ra^{228}(6.7 yrs); Th^{228}(1.9 yrs); Ra^{224}(3.64 days)-Thoron^{220}(54.5 sec); Bi^{212}(60.5 min); Po^{212}(very short); Tl^{208}(3.1 min). The decay chain of thorium contains isotopes which emit particles and photons of much greater energy than uranium; it is, therefore, more hazardous than uranium.

Fission Products of U^{235}, U^{233}, or Pu^{239} Fission

The fission products resulting from any of the three fissionable isotopes are generally similar in their distribution between the elements starting with atomic number thirty (Zn^{72}), and atomic number sixty-six (Dy^{161}). From the standpoint of waste disposal, those of most interest possess long half-lives and/or have high hazard potential for man as defined by their allowable concentrations in the human body.

Those most significant, classified by half-life and yield, are:

	Half-life	Approximate % of Total FP Activity after Decay		
		100 days	3 years	30 years
Cs^{137}	26.6 yrs	<2	15	~49
Sr^{90}	28.0 yrs	<2	15	~49
Pm^{147}	2.6 yrs	3	15	<1
Ce^{144}-Pr^{144}	290 days	45	50	-
Kr^{85}(gas)	10.3 yrs	<1	1	<1
I^{131} (gas)	8.1 day	<1	-	-
Zr^{95}-Nb^{95}	63 days	33	-	-
Ba^{140}-La^{140}	12.8 days	<1	-	-
Ru^{103}-Rh^{103}	41.0 days	5	-	-
Ru^{106}-Rh^{106}	1.0 yr	2	3	-
Sr^{89}	54 days	7	-	-
Xe^{133}(gas)	5.27 days	<1	-	-

Note: Irradiation basis is exposure for one year to 3 x 10^{13} thermal neutron flux.

The fission products are the major new radioactive materials produced by the fission process; they appear in small concentrations in many waste streams in the fuel cycle, but to the greatest extent in the waste streams from reactor fuel reprocessing plants. For the long periods over which we must consider storing radioactive wastes, it is apparent that the fission products cesium-137 and strontium-90, along with the noble gas, krypton-85, are of most importance.

The energy released by fissioning of uranium or plutonium amounts to approximately 201 Mev per atom fissioned, or approximately 24,000 kw (1 megawatt day, if fissioned at the rate of 1 g/day) of heat per gram of material fissioned. The major portion of this - over 80 per cent - appears as heat generated by fast moving fission fragments. Part of the remaining 20 per cent - approximately 6 per cent - of the energy is released slowly by the decay of radioactive fission products. The fission products decay with emissions of beta particles and gamma rays. The rate of energy emission decreases rapidly at first, more slowly later as shown by the following, taken for long irradiation (350 days) at a neutron flux of 3 x 10^{13} $n/cm^2/sec$:

(1) A total fission product power of 38.5 watts/gram of U^{235} originally is available at time of shut-down.

(2) Decay reduces this rate by a factor of 10 in three hours to 3.85 watts/g U^{235} originally

(3) There is a factor of 100 reduction in heat release in 58 days; a factor of 1000 in 520 days; a factor of 10,000 in 4050 days.

Radioactive Elements Produced by Parasitic Neutron Capture

Neutrons produced during the fission process interact with and ultimately are captured by the surrounding atoms in a reactor. The four main processes in which neutrons (produced on the average of 2.5 fast neutrons per fission for U^{235}-3.0 per fission for Pu^{239}) take part are:

(1) Complete loss of neutrons from the system; this loss is kept to a minimum by efficient reactor design.
(2) Nonfission capture by fissionable and fertile nucleii like U-235, U-238, Th-232 or Pu-239.
(3) Nonfission capture - referred to as parasitic capture - by the moderator, coolants, structural materials, fission products, and impurities in the fuel moderator and coolants.
(4) Fission capture of neutrons of various energy levels by fissionable materials and of fast neutrons by U-238 and Th-232 if they are present.

For a typical thermal reactor, using natural or slightly enriched uranium, the neutrons would be directed by design into the following approximate distribution: (4)

100 (fast) neutrons produced by fission
↓ → 5 leak out during slowing down
↓ → 10 absorptions (resonance) in U^{238}
85 (slow) neutrons produced
↓ → 5 neutrons available for absorption
80 (slow) neutrons available for absorption
↓ → 33 absorbed by U^{238}, other heavy elements, moderator, poisons
↓ → 7 nonfission capture by U^{235}
40 absorbed by U^{235} to cause fission and to produce fission products
↓
100 fast neutrons, to sustain the cycle

The neutrons absorbed by nonfission processes produce in part, other radioactive nucleii which must be considered as contributors to the radioactivity of wastes from the fission-fuel cycle. The most important of these parasitic captures from the waste disposal standpoint are those which:

(1) Activate the circulating coolants and coolant impurities in a reactor

(2) Activate the structural materials of the reactor and fuel elements, and corrosion products which circulat in the coolant to the neutron field

(3) Are absorbed by nonfission capture in fissionable and fertile isotopes to produce new heavy elements that have long half-lives and will appear in some very small but significant amounts in the fuel recycle and in the waste streams from radiochemical separations plants.

Obviously, the neutrons that are absorbed by fission capture in U^{235}, U^{233} or Pu^{239} (or fast fission for U^{238} and Th^{232}), produce the major radioactive constituents in wastes, the fission products.

Description of Wastes from the Nuclear Power-Feed Preparation-Fuel Recycle

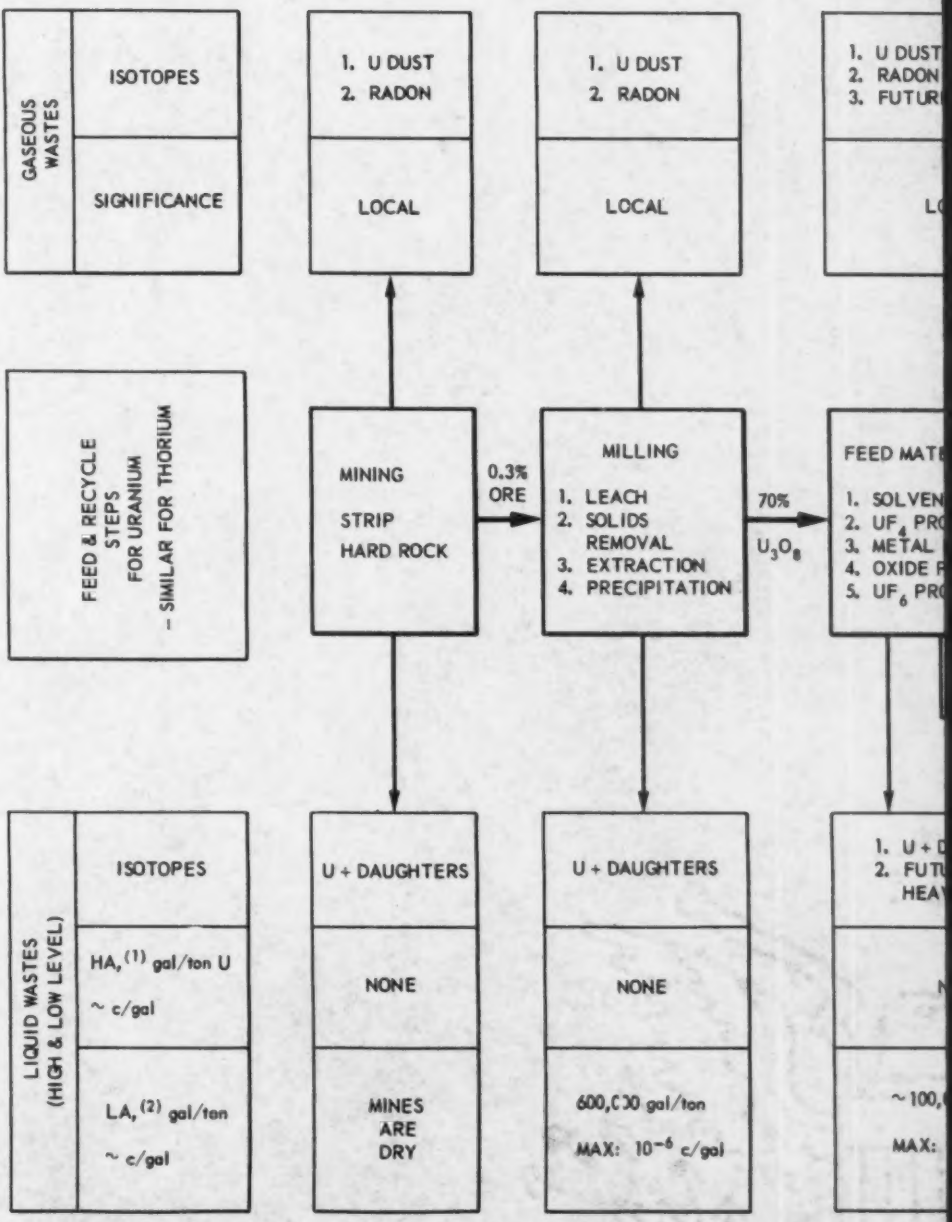
There are many separate operations which are necessary to prepare fissionable and fertile materials as satisfactory reactor fuels; there are several general classes of materials which must be brought together in proper balance to operate a self-sustaining, heat-producing reactor; and there are many steps required to recycle and recover fissionable and fertile material from irradiated fuel. Each of these produces solid, liquid, and gaseous wastes in which the concentration, half-lives, and relative hazards of contained radioactive materials varies. A general flowsheet, with notations concerning the type of waste, for the nuclear fuel cycle complex is given in Figure 1.

Discussions of Wastes from Raw Materials Production Operations

(Data taken from references (5)(6)(7)(8) which are to be part of the published record of these hearings)

Uranium is recovered from the product of many mines and 26 mills in the western United States, plus very small operations in the Florida leach zone area. The western mills range in size from 50 to 3500 tons of ore per day, with a total daily mill feed capacity of approximately 21,000 tons of ore per day. For all mills the average uranium content is approximately 0.25% expressed as U_3O_8 (multiply by 0.85 to get elemental uranium). Thus, the daily production is approximately 50 tons of U_3O_8 concentrates per day as impure oxides. This production is approximately 35 per cent of the total non-Communist bloc product of uranium.

The waste products of uranium mines are not significant from the general waste disposal picture. The principal problems are radon gas, protection of miners from



(1) HIGH ACTIVITY WASTE: ACTIVITY IS ORDER OF MAGNITUDE ESTIMATE, CURIES/GAL
(2) LOW ACTIVITY WASTE;

FIGURE 1. SC

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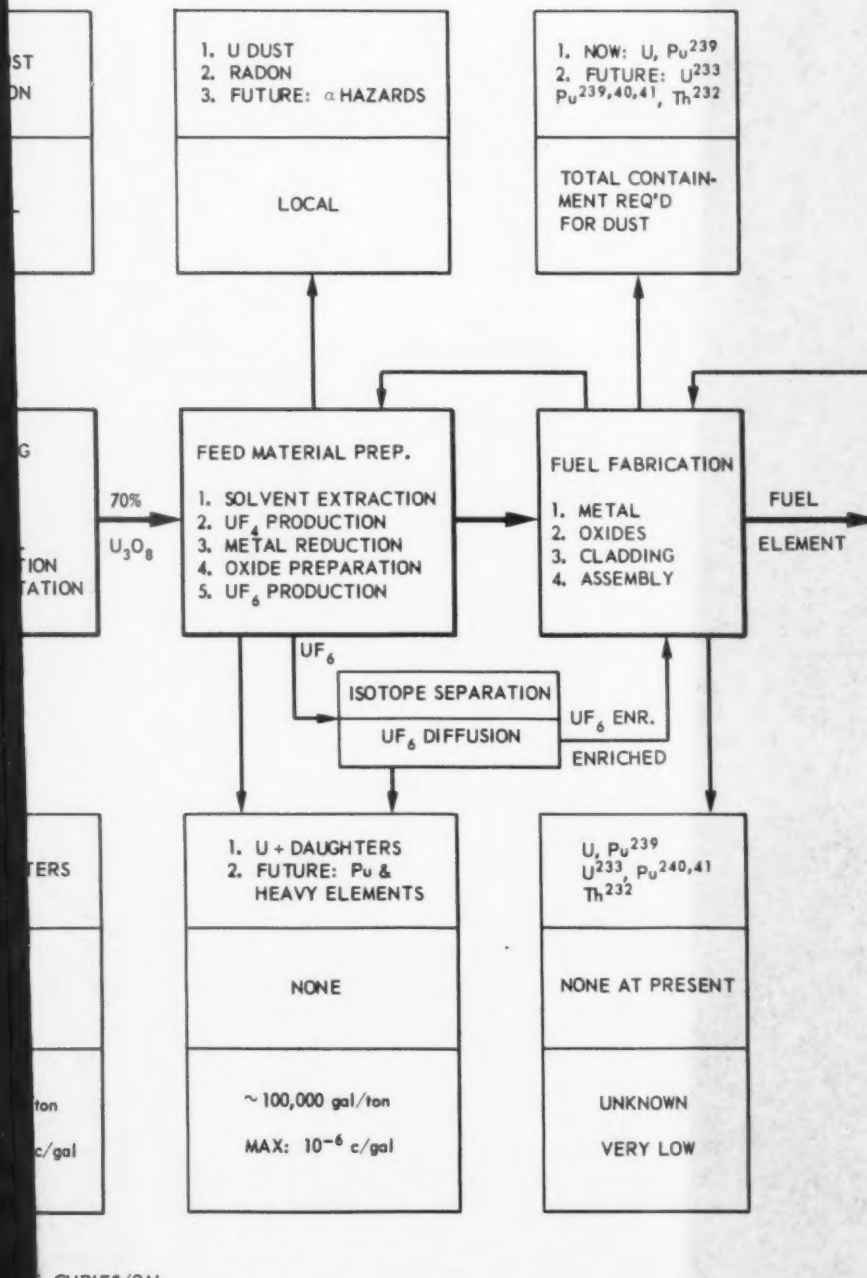

, CURIES/GAL

FIGURE 1. SCHEMATIC FLOWSHEET FOR ORIGIN OF WASTES

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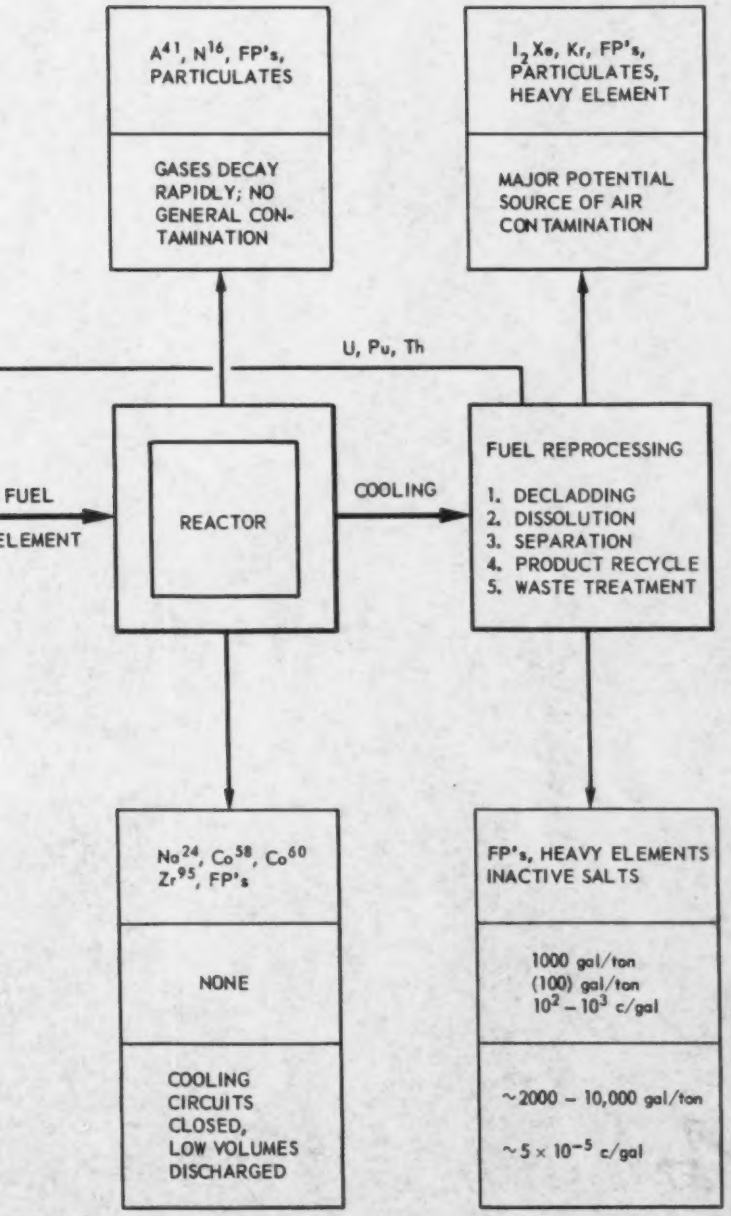
ORNL-LR-DWG 35861



radioactive dust, and a decay product of uranium which is released by the uranium-bearing minerals, mostly carnotite or autunite. Harmful concentrations of these airborne radioactive materials are prevented by sufficient ventilation. The air exhausted from the mines does not present a problem of general atmospheric contamination.

The problem of handling slightly radioactive water from most of the mines is non-existent because most of the mines are dry. In the few areas where mine water pumping is required, such as the Ambrosia Lake, New Mexico area, the water which is removed presents no disposal problem.

The uranium mills produce several types of wastes which must be disposed of to the environment. The most significant problem for the area immediately adjacent to the mills is that associated with the large volumes of process raffinates which are produced. The production rate of contaminated water from milling operations is approximately 865 gallons per ton of ore treated, or 285 gallons of liquid waste per pound of uranium (not as U_3O_8) produced, for a total daily production of 18 million gallons. This quantity of water, minus about 60 gallons per ton which remains with solid tailings, is disposed of by overflowing large retention ponds to a stream, if available, or allowed to percolate into the ground.

The total quantity of radioactive material in the uranium ore as delivered to the mills is less than 100 curies per day. Most of this appears in the liquid tailings which contain large quantities of chemicals, some of which also present problems in disposal. The following Table 1 summarizes preliminary studies on radioactive contamination of the mill wastes:

Table 1

Isotope	Selected Measurements of Activity micro curies/ml	Permissible concentration for unrestricted areas[2] micro curies/ml	Fraction of Permissible Level
1. U natural	3×10^{-5} to 9×10^{-8}	7×10^{-6}	max. 40
2. Radium-226	2.7×10^{-6} to 8×10^{-8}	4×10^{-9}	all over
3. Thorium-230	3.9×10^{-5} to 2.8×10^{-8}	not given	
4. Thorium-234 plus Pa^{234}	1.2×10^{-4} to 1.8×10^{-7}	3×10^{-1}	all under
5. Po^{210}	not reported	3×10^{-6}	
6. Radium-222	not reported	2×10^{-7}	

Note: Two measurements out of six are slightly high. These can be controlled easily in recovery process by adjusting acid strength in tailings effluent.

The problems of disposal of wastes from milling operations at the moment present severe local problems which affect water supplies in an area where water is in short supply. The concentration of radioactive material in the effluent surely can be reduced to tolerance levels at additional costs to the product uranium. Here for the first time, we have an example of one of the most difficult of the general waste problems, that of handling large volumes of wastes containing very small, but above tolerance, concentration quantities of radioactive material.

Radioactive wastes in the mining and milling of thorium ores are of a similar nature to those for uranium, but the radioactivity of the wastes is a more serious problem because of the presence of the more energetic decay products of thorium and their lower biological tolerances. Most of the thorium produced to date has been recovered from monazite sand which is primarily from India, Brazil, Australia, and to a very limited extent, from North Carolina and Florida stream beds. If appreciable quantities of thorium are recovered in the United States from imported or domestic ores, the control of wastes from these operations will require close attention. Sizeable production of rare earths, which occur along with thorium in the weathered pegmatite as part of the primary mineral, monazite, also produce wastes of a similar nature. Thoron gas is more radioactive than its counterpart, radon, in the uranium decay series because of its shorter half-life and its slightly more energetic alpha particle.

Since thorium has not yet become a major fuel material, the problems peculiar to its recovery from ore will not be discussed further.

Feed Materials Operations

Ore concentrates containing on the average of 70 per cent U_3O_8 are further chemically purified to produce uranium salts that are fed to the gaseous diffusion cascades for the separation of U^{235} from U^{238}; reduced with Ca or Mg to uranium metal; or converted to uranium compounds such as uranium dioxide to be used as power reactor feed material. The refining of ore concentrates, both domestic and foreign, is accomplished by solvent extraction processes (with the exception of the new fluoride volatility plant of the General Chemicals Company) in integrated centers by Mallinckrodt of St. Louis and Weldon Springs; and by The National Lead Company of Ohio at Fernald, Ohio. Purified uranium is produced as a nitric acid solution which is converted through a series of chemical steps to uranium tetra-

fluoride. This salt is then converted to uranium metal for power reactor or plutonium production reactor use.

The radiochemical nature of the liquid wastes from the processing of ore concentrates is similar to that of the mill raffinates, containing natural uranium and its decay daughters. For each ton of uranium processed, approximately 1000 gallons of solvent extraction raffinates are produced; if recycled uranium from within the feed materials complex is added to this, the volume of purification cycle raffinates may be on the order of 1500 gallons for each ton of fresh concentrate introduced. Because this waste is low in salt content it is volume-reduced and stored; only condensate below acceptable tolerance is discharged.

Additional volumes of liquid wastes containing only minor quantities of radioactivity result from operations leading to UF_4 product. Gases from the reduction and hydrofluorination of UF_4 must be scrubbed to remove chemical fumes and entrained uranium. Scrubber effluents are monitored for activity level and discharged to streams or rivers at below tolerance levels.

Waste materials from uranium metal production are leached to remove contained uranium; the leach liquors are treated by the solvent extraction purification cycle or by ion exchange. Raffinates from these operations are discharged at safe levels to the environment. The solid reduction bomb liner - MgF and fused dolomite - after leaching, is sufficiently free of uranium and radioactive materials to be marketed as a possible raw material for Mg production.

Diffusion plant feed can be prepared from two sources of uranium: (1) concentrates from milling; and (2) recycled uranium from production reactors, such as those at Hanford and Savannah River, or the Idaho Chemical Processing Plant. The uranium product from large radiochemical separations plants has been allowed to decay for sufficient time to permit the decay of parasitically-produced U^{237}, which is a beta-gamma emitter with a 6.7 day half-life resulting from a double neutron capture by U^{235}. The uranium recycled from the reactors is separated from Pu so that there are only 10 parts of Pu per billion parts of U. Normally, the fission products are removed completely, as well as the decay daughters of uranium itself. Wastes from the gaseous diffusion complex are carefully treated, monitored, and discharged at below tolerance levels.

Estimating the volume of slightly contaminated water which is discharged from a feed materials plant is difficult; it has not been possible to relate volume to the number of tons of uranium processed. During 1955, it has been reported that approximately 95,000,000 gallons of treated water was discharged to the surface at the Feed Materials Center, Fernald, Ohio.(9)

As is the case with the liquid wastes from milling of uranium ore, the feed material liquid wastes are characterized by large volumes of liquid containing below tolerance concentrations of radioactive materials.

Solid wastes, such as uranium-contaminated ore drums; used equipment and piping; filter cakes from scrap and MgF_2 and dolomite liners for metal reduction; solid residues from various operations; are easily disposed of by burial in protected areas.

Radioactive Wastes from Operating Reactors

Radioactive materials, other than those occurring in nature, are generated in a nuclear reactor. The nonfission capture of neutrons by coolants, impurities in coolants, structural materials and corrosion products produces radioactive material which must ultimately be discharged from the reactor. Fission products can be released from ruptured fuel elements either as a single burst or gradually, depending upon the nature of the rupture. Fission products and new heavy elements, produced by fission and fertile or heavy element absorption of neutrons, usually remain sealed in fuel elements until they are dissolved in radiochemical processing plants.

Active Material in Water Cooled Reactor Cooling Circuits

In water cooled reactors the radioactive nuclides produced by the irradiation of water - not considering impurities and corrosion products - produce radioactive nucleii of short half-life, but of sufficient radioactive intensity to control the thickness of shield which protects the cooling circuit pumps and piping. From water, the following are produced by (n,γ) or (n, p) reactions: nitrogen-16 (half-life: 7.4 sec); nitrogen-17 (half-life: 4.14 sec); oxygen-19 (half-life: 29 sec). Additional nitrogen-16 and nitrogen-17 can be produced from dissolved oxygen and nitrogen in water; most pressurized water reactors, therefore, use deaerated water. The minor amount of argon present in air, which can be dissolved in cooling water will capture a neutron to produce argon-41 (half-life: 1.82 hrs). These gases have short half-lives and although of importance

in the design of reactor facilities, they are of no consequence for waste disposal consideration except where once-through cooling is employed, a practice that is uneconomic for power producing reactors.

The long-lived activity in water cooling systems results primarily from the activation of structural or fuel components in-core with subsequent release to the coolant; or from corrosion products which are swept into the neutron field by the coolant. Some of the irradiated products are deposited in the external cooling systems of the reactor as indicated by the heat exchanger radiation levels in the S-1-W reactor[10], which were 600 mr/hr at full reactor power and 350-400 mr/hr at 10 per cent full power. The principal constituents contributing to this activity were measured after 93 full-power hours of operation; similar measurements were made with the Army Package Power Reactor coolant after 700 full-power hours. The nuclides identified in solids for the S-1-W, and in solids plus circulating water for the APPR-1 were:[10]

Nuclide	Half-life	S-1-W (Zirconium) % of Total	APPR-1 (Stainless Steel) % of Total
Co^{60}	5.2 years	19.5	55.4
Co^{58}	71 days	5.5	-
Cr^{51}	27 days	1.1	0.4
Zr^{95}	65 days	0.7	-
Fe^{59}	45 days	0.7	7.3
Hf^{181} (175)		53.3*	-
Mn^{56}	2.6 hours	in solution	14.2
% Identified		80.8	77.3

*from abrasion of hafnium control rod

Other nuclides of importance that might be present in water coolants are:

Na^{24}	15 hours	produce from Al^{27} in systems utilizing aluminum
Fe^{55}	2.9 years	produced from Fe^{54}

From the standpoint of waste disposal the most significant nuclides are Co^{60} and Fe^{55} because of their longer half-lives. For reactors used for plutonium production, water is pumped once through only; the resulting activated water and

water impurities are allowed to decay for a period of hours, which reduces most of the activity to tolerable discharge levels.

Circulating water systems are purified using ion exchange resins and associated filters on side streams.

Little quantitative data exists to relate radioactive material discharge with discharge coolant volumes and reactor operating power. Experience at Argonnel National Laboratory with the Experimental Boiling Water Reactor (20 Mw thermal power) for one year of operation is summarized:[11]

EBWR - 127 cu. ft. solid waste containing average of 90 millicuries/cu. ft.
- 200,000 gallons of water with an average specific activity of 2×10^{-4} millicuries/gallon, for a total of 11.4 and 40 curies, respectiv

In addition to liquid wastes and solid wastes, radioactive gases are released. In many reactors shielding, instrument tubes, and other auxiliaries are cooled by air. Argon-40, present as an impurity in air, captures a neutron to become Argon-41 (half-life: 1.8 hrs), which presents no waste disposal problem because of its short half-life.

To summarize, induced activity in cooling water circuits provides another source of large volume low activity wastes. The radionuclides can be removed by ion exchange and filtration.

Activity in Other Coolants

The induced activity in liquid metal reactors is very high because of the production of Na^{24} However, sodium is not removed from the reactor coolant system so that the activity is contained or can be allowed to decay before discharge.

Organic cooled and moderated reactors produce reasonably large quantities of organic residues and gas as a result of the radiation decomposition of organic coolants such as the di- or terphenyls. Based on limited experience with the Organic Moderated Reactor Experiment, a 100-Mw reactor might produce 300 cubic feet of nonradioactive gas (except for fission gas leaks from ruptured fuel), and 3600 pounds of essentially nonradioactive solid sludge.[12]

Origin of Concentrated Fission Product Wastes (Prepared by J. O. Blomeke, et al (13) Oak Ridge National Laboratory

Irradiated reactor fuels are chemically processed to reclaim the unburned nuclear fuel and recover the transmutation products, such as Pu^{239} or U^{233}, from mixtures of the fission products and inert components of the fuel. At the present time, these are separated by solvent extraction. While the processing details vary with the fuel being processed and the plant performing the operation, the basic principles are the same. Solid fuels, which have been stored 90-120 days following removal from the reactor (to permit decay of short-lived fission product activities), are dissolved in nitric acid, and the solution is fed to the mid-section of an extraction column where it is contacted with an immiscible organic solvent that enters at the bottom. The solvent selectively extracts the uranium and plutonium; and an aqueous salt solution, added to the top of the column, scrubs the traces of fission products from the organic extract before it leaves the top of the column. The waste, containing more than 99.9% of the dissolved fission products, and inerts in the feed and scrub solutions, leaves at the bottom of the column. The uranium and plutonium are separated from each other and decontaminated further from fission products by additional treatment; but the waste from these steps contains very small amounts of activity and, in some cases, is recycled to the process.

Several processes in current use for processing irradiated fuels are shown in following table. It should be noted that the volumes given refer to the wastes as they come from the extraction column. In current practice, the wastes are evaporated to smaller volumes and are frequently neutralized before being stored in underground steel and concrete tanks.

Characteristics of Solvent Extraction Processes

Process	Application	Solvent	Salting Agent	Approx. Vol. Untreated High-activity Waste
Purex	Pu-natural Uranium	TBP in hydrocarbon	HNO_3	990 gal/metric ton U
Redox		Hexone	$Al(NO_3)_3$	1000 gal/metric tonU
Hexone-25	Enriched U-Al alloys	Hexone	$Al(NO_3)_3$	700 liters/kg enriched U
TBP-25		TBP in hydrocarbon	HNO_3 + $al(NO_3)_3$	670 liters/kg enriched U
Thorex	$Th-U^{233}$	TBP in hydrocarbon	$Al(NO_3)_3$	1360 gal/metric ton Th

Characteristics of Present Wastes

Three types of irradiated fuels are currently being processed: (1) natural uranium, where plutonium and uranium are separated from fission products and from each other; (2) uranium-235 - aluminum alloys where uranium-235 is recovered and separated from fission products; and (3) thorium, where uranium-233 and thorium are separated from fission products and each other. In each process the fuel is dissolved with nitric acid solutions, in preparation for the subsequent liquid-liquid extraction step. The solvents used for extraction are tributyl phosphate (TBP) or methyl isobutyl (hexone). When TBP is used, it is diluted with a paraffinic hydrocarbon in order to obtain more favorable physical characteristics, and nitric acid is usually used as the salting agent in the aqueous scrub. This yields a waste (see below) that can be evaporated to much smaller volumes than is possible when aluminum nitrate is used as a salting agent, as in hexone extractions.

Table I gives the characteristics of the high-activity wastes produced by these processes. It should be emphasized that the numbers given are all approximations, the actual values depending on the operating characteristics of the particular chemical plant and the details of the chemical flowsheet being used. In addition to the concentrations of inert chemicals given here, all the wastes contain fission products and lesser amounts of uranium, plutonium, and other heavy elements. The presence of these heavy elements is attributable mainly to process losses which usually approximate 0.1%.

Purex-type wastes are mainly solutions of fission products in nitric acid and their physical properties are essentially those of about 1 M HNO_3. Trace constituents are iron, nickel, and chromium, present in varying amounts as products of corrosion of the stainless steel process equipment. Because of their low solids content, these wastes can be concentrated by evaporation to relatively small volumes and much of the nitric acid distilled and recovered for reuse in the process. Purex-type waste can be concentrated to about 8 M HNO_3, at which point oxidation of the fission product ruthenium to the volatile RuO_4 begins. The waste is then neutralized by addition of 50% NaOH solution and evaporated to near the solubility limit of the dissolved salts. Recent work at Hanford has reduced the fission product raffinate volumes to about 80 gallon per ton of U.

Table 1

CHARACTERIZATION OF FIRST CYCLE HIGH LEVEL AQUEOUS WASTES FROM SELECTED SOLVENT EXTRACTION PROCESSES

Process	Chemical Properties, Constituents (M), Approximate, Exclusive of Fission Products and Heavy Elements																Waste Volume, gal/g U^{235} Consumed	Approximate Specific Gravity	Approximate Concentration of U in Feed g/liter	Waste Activity(2)		
	H	Al	Fe	Cr	Ni	Zr	Na	NH_4	NH_3	Gd	Mn	Mg	Hg	NO_3	F	SO_4	PO_4	Cl				curies/gal	Total watts/gal(3)	BTU/hr/gal
Redox	-0.2	1.2	•	•	•		0.46							4.1					0.27	1.18	450	1720	8.6	29.4
Purex	2.5 to 7.0		•	•	•		•							2.5 to 7.0		•			0.34	1.07	300	1320	6.7	22.7
Thorex	-0.1	0.5	•	•	•		•						0.01	1.4	0.05		•		0.34	1.12	350 (Th)	81	0.4	1.37
Hexone-"25"	-0.2	1.5	•	•	•			0.5					0.01	4.5					0.31 to 0.13	1.23	2-5	1620 to 3940	8.1 to 20.2	27.7 to 69.0
TBP-"25"	0.5	1.6	•	•	•								0.01	5.5		0.02	•		0.22 to 0.11	1.29	3-6	2580 to 5160	12.9 to 26.0	44.1 to 88.6
Zirconium-HF for enriched U	0.8 to 2.0	.8		0.1		0.5				0.02				2.3	3.0				~1.50	~1.1	0.3	350	1.75	6.0
Stainless Steel-H_2SO_4 for enriched U-235	2.6	0.25	0.05	0.01	003		•				0.001			2.4		0.30			~0.35	~1.3	2-3	1500	7.5	25.0

Notes: (1) Wastes are untreated; they are essentially as they leave the solvent extraction plant and are subject to further treatment such as evaporation, neutralization, chemical treatment for fission product removal, etc.

(2) Basis for activity numbers: Irradiation period 4000 Mwd/t for natural uranium
5×10^{13} n/(cm)^2(sec)
4000 grams U^{233} chain per ton of thorium
53% burn-up for U^{235} in enriched fuel elements
100 days decay cooling from time of reactor discharge

(3) After 100 days' decay, the distribution of energy is approximately 50% γ and 50% β.

(4) Waste volume per gram U^{235} consumed is an inverse function of burnup; i.e., for hexone-25 at 20% burnup, the gal/g $U^{235} = (\frac{53}{20})(0.14)$.

(5) Waste activity varies approximately as the (irradiation level)$^{0.2}$

The wastes from the rest of the processes considered here contain significant amounts of aluminum. In the Redox process, the aluminum is added as a salting agent. In the processes for enriched uranium, it is present in the feed solution since it is a major component of the original fuel. In the Thorex process, it is used to provide what is known as an "acid deficient" condition, i.e., one in which the aluminum partially hydrolyzes to the monobasic or dibasic nitrates. The acid deficient solution is also used in the Redox and Hezone-25 processes.

The Redox waste contains some sodium, which is added in adjusting the feed solution to an acid deficient condition, and some dichromate, which is used to oxidize the plutonium to an extractable valence state. Ammonia is used to adjust for acid deficiency in the TBP-25 process.

The Hexone-25 and TBP-25 wastes have mercury, and the Thorex waste has both mercury and fluoride, which are added as catalysts in dissolving the fuel.

Ferrous sulfamate is added in the TBP-25 process scrub solution to reduce plutonium to a nonextractable valence state, thus effecting its separation with the fission products from uranium.

The trace constituents in these wastes include iron, nickel, and chromium as products of stainless steel corrosion, and, in some cases, small amounts of silicon and other impurities originally present in the fabricated fuel. Phosphate and sulfate are used in the Thorex process to increase the decontamination of specific fission products and remove interferring ions.

The aluminum-bearing wastes are normally evaporated until the aluminum concentraction reaches 2.2 M. If neutralization is desired, the pH is adjusted to about 13 by addition of 50% NaOH solution, 25-30% in excess of the stoichiometric amount required to form sodium aluminate.

High level wastes now in storage at the major processing plants are given in the following[15]

	Total
Wastes in storage (10^6 gallons)	72.7
Total tank capacity (10^6 gallons)	103.8
Total number of tanks	170

Fission Product Levels

Although the fission products are the source of most of the radiation hazard, they do not greatly influence the chemical properties of the wastes because they are present in such low concentrations. For every mega-watt-day of irradiation, about 1.1 g of fission products--both radioactive and stable--is formed. On this basis, it can be seen that most of the untreated wastes contain of the order of 1 g/liter of total fission products, consisting of isotopes of some 37 elements lying between zinc and dysprosium in the periodic table. While the total number of fission product atoms does not change after irradiation, the concentration of specific fission products does change with time, and for any given fuel is a function of the reactor power level during irradiation, the length of time the fuel has been irradiated, and the time elapsed since discharge from the reactor. The estimate of individual fission product levels and the nuclear properties of fission product mixtures is beyond the scope of this paper, but an extensive correlation of such information for thermal fission of U^{235} has been published.(14)

The contribution to the total beta activity by fission products of ten different mass numbers is presented in Figure 2. For the purposes of this illustration, natural uranium irradiated in a thermal neutron flux of 3×10^{13} $n/cm^2/sec$ for one year was chosen as a basis. At a decay time of 10 days, 63 d Zr^{95} and its daughter, 35 d Nb^{95}, account for 20 percent of the total beta activity. They are followed in order of decreasing significance by 12.8 d Ba^{140} and 40 h La^{140} and by 290 d Ce^{144} and its 17.5 m Pr daughter. In the range of 100 to 1000 days decay, 290 d Ce^{144} - 17.5 m Pr^{144} represent 40 to 60 percent of the total activity. After this time, Sr^{90} and Cs^{137} and their daughters begin to assume an increasingly dominant role. In all, the isotopes considered here represent about 63 precent of the total activity after 10 days decay, 87 percent after 100 days decay, and greater than 98 percent at 1000 days and thereafter.

One of the properties of fission product mixtures of most interest to waste disposal is their heat emission. In any disposal scheme, provision must be made to dissipate the heat evolved as a consequence of radioactive decay if elevated temperatures which can lead to chemical instability, corrosion, and entrainment during storage are to be avoided. A typical curve of the fission

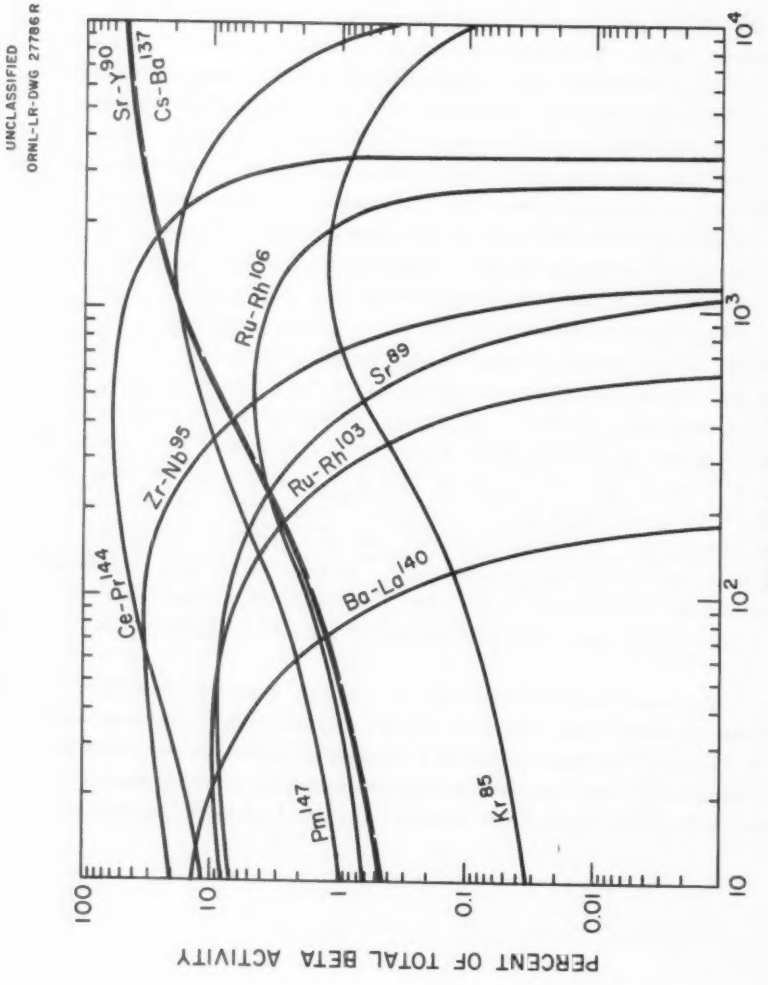


FIGURE 2 DISTRIBUTION OF FISSION PRODUCT ACTIVITY AS A FUNCTION OF DECAY TIME. BASIS: IRRADIATION AT THERMAL FLUX OF 3×10^{13} n/cm²/sec FOR 1 YR

product power expected in wastes from processing a fuel of high burnup is given in Figure 3. In this figure the specific fission product power (in watts/gal of waste) is plotted against the time of decay. If it is assumed that 800 gal of waste has been produced from processing 1 metric ton of natural uranium, irradiated to 10,000 Mwd/ton at a specific power of 33 Mw/ton, besides the total power, the contributions are noted for the rare gas fission products (normally separated when the fuel is dissolved), the rare earths, which are the major heat producers, and the long-lived hazards, Cs^{137} and Sr^{90}. After one day's decay, the fission power is rather formidable, 1700 watts/gal; but after a 100 days' decay, a reasonable time to allow before processing this fuel, the power has dropped to about 200 watts/gal. The power emission continues to fall until a decay time of about 3300 days, or nine years, the rare earths have begun to drop to a level where strontium and cesium and their daughters constitute the main source of power evolved in the waste. At this point the total power amounts to about 0.5 watt/gal and decreases according to the 27 to 28-year half-life of Sr^{90} and Cs^{137}.

FIGURE 2 DISTRIBUTION OF FISSION PRODUCT ACTIVITY AS A FUNCTION OF DECAY TIME. BASIS: IRRADIATION AT THERMAL FLUX OF 3×10^{13} n/cm^2/sec FOR 1 YR

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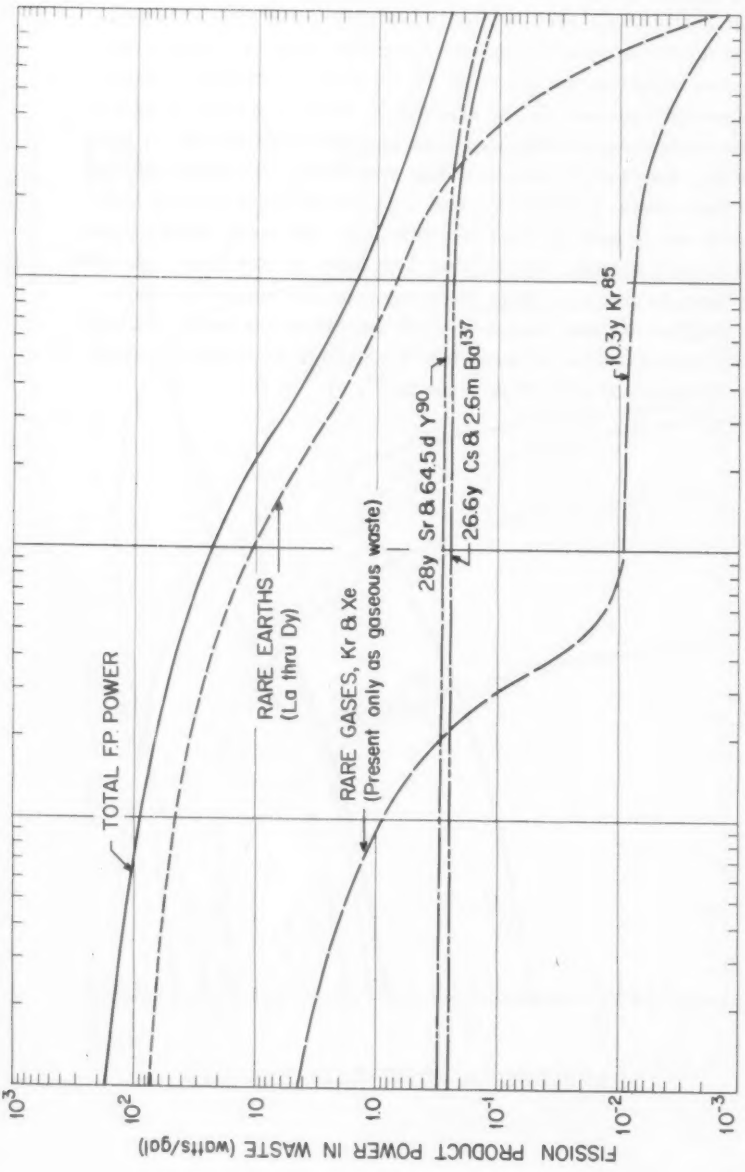
FISSION PRODUCT POWER IN WASTE (watts/gal)
TIME OF DECAY (days)
TOTAL FP POWER
RARE EARTHS (La thru Dy)
RARE GASES, Kr & Xe (Present only as gaseous waste)
28y Sr & 64.5d Y90
26.6y Cs & 2.6m Ba137
10.3y Kr85

FIGURE 3 FISSION PRODUCT POWER IN WASTE AS A FUNCTION OF DECAY TIME. BASIS: 800 GAL WASTE PER METRIC TON U, IRRADIATED 10,000 Mwd/ton At 33 Mw/ton SPECIFIC POWER

FIGURE 3 FISSION PRODUCT POWER IN WASTE AS A FUNCTION OF DECAY TIME. BASIS: 800 GAL WASTE PER METRIC TON U. IRRADIATED TO 10,000 Mwd/ton At 33 Mw/ton SPECIFIC POWER

Comments on High-Level Wastes from Possible Processes other than Solvent Extraction

At the present time, and probably for some time to come, all radiochemical fuel reprocessing is being done by organic solvent extraction; the existing solvents will be used to recover fissionable and fertile material from feed solutions prepared by a variety of aqueous feed preparation methods, each adapted to a specific type of power reactor fuel. The solvent extraction processes are likely to remain major techniques because of their versatility and economy. It is probable that a single reprocessing plant will serve many power reactors; if 12,000 Mwd/metric ton (heat) burn-up can be achieved, reprocessing costs in such a centralized processing plant with a capacity of 6 tons/day will probably be about 0.3 mil/kwhr (electricity). Such a central plant would process the fuel output of possibly ten large power reactors.

New processes are being developed. For reactors of high fuel and fertile material content, such as the fast breeder, where inventory charges are high, processes which give partial fission product decontamination are being studied. The pyrometallurgical processes being studied at Argonne for EBR-II are examples of this type of process. Wastes from the fuel melting slagging, or salt washing, will be produced principally in solid form. The exact nature of wastes from this kind of process are not yet known.

Another process techniques which has been operated successfully on a large pilot plant scale (as yet, still at low activity levels) at Oak Ridge National Laboratory, and on a smaller scale at Argonne is the fluoride volatility process. Fuel elements will be dissolved in a fused fluoride salt mixture with hydrofluoric acid gas at temperatures of approximately 650°C. The uranium can then be removed from the fused salts and most of the fission products with fluorine gas, which converts UF_4 in the fused salt to volatile UF_6 gas. The few fission products that follow the uranium can be removed by absorption on sodium fluoride or by distillation. The fission products would be discharged as a component of the solidified salt mixture. This process has yet to be demonstrated on an engineering scale. It also now is limited in application to fully enriched U^{235} fuels only.

Problems of Active Gaseous Wastes from Radiochemical Plants

The fission product gases which remain after the normal cooling period of approximately 100 days are primarily Kr^{85}, a noble gas with a 10.27 years' half-life; and I^{131} in small quantities because of its relatively short half-life of 8.0 days. If processing reactor fuel elements is started with a shorter decay period than 100 days, iodine-131 becomes a controlling isotope in the gases which are released during dissolution operations. Xe^{133}, a noble gas with a half-life of 5.27 days, will also be present in significant concentration in dissolver off-gas streams.

Iodine must be removed completely from the off-gas streams; processes exist that are 99.99% efficient for I_2 removal which have been operated for many years. Dissolver gases contain, along with the fission gases and air, gaseous oxides of nitrogen and possibly other acid gases. Iodine is removed from the waste gas by passing it over a bed of silver nitrate coated ceramic packing at elevated temperature. The oxides of nitrogen are removed by scrubbing with sodium hydroxide; krypton and xenon, if present, may be diluted with air and discharged through tall stacks to the atmosphere. Both xenon and krypton can be removed by adsorption on charcoal at low temperatures; by liquefaction; or by other methods currently under investigation.

If natural uranium is irradiated in an average flux of 3×10^{13} $n/cm^2/sec$ for a year, there will be the following concentrations of the above isotopes:

Isotope	Half-life	curies/g U^{235} consumed at 20 days decay	curies/g U^{235} consumed at 100 days decay
Kr^{85}	10.27 years	0.358	0.354
I^{131}	8.0 days	10.3	0.0109
Xe^{133}	5.27 days	11.8	0.00110

Small particles which contain radioactive elements present very serious problems. From the radiochemical plant it is possible to release aerosols of high specific activity; many operations in the chemical process produce mists or entrained solids, such as solution transfer by steam jet, evaporation, and mixing with air sparging. All chemical plants must be provided with filters of exceptional efficiency. Particles containing highly alpha-active materials, such

as mixtures of the plutonium isotopes, thorium and uranium-233, protactinium-233, and the heavy element produced by parasitic neutron capture by fissionable and fertile isotopes.

The one additional fission product that has appeared in very high concentrations from certain steps of the solvent extraction processes is ruthenium. Under the strongly oxidizing conditions required for a scavenge precipitation step to remove ruthenium, zirconium, and niobium as solids on manganese dioxide, part of the ruthenium oxidizes to the volatile RuO_4, which then plates out as a solid scale on vent lines and other equipment, or passes through the equipment and filters to condense later as very fine particles of high specific activity. The loosely attached films containing ruthenium flakes off, and appear as highly active particles; this is a particularly difficult problem during maintenance periods when equipment and piping are changed. If this problem is experienced in a process, there are known conditions to prevent or correct it for the distillation of ruthenium oxide.

Low and Intermediate-Level Liquid Wastes
(Excerpt from reference (15)

Low-level or intermediate liquid wastes result from fuel processing operations also, usually from second and third cycle separations and from fuel dejacketing. In addition, low-level liquids derive from reactor cooling systems, laboratory operations and laundry operations. In proper environmental situations, these wastes are susceptible to either direct dispersal to the environment or discharge following treatment. Because of their large volume, millions of gallons per day, total costs of treatment, where required, tend to be substantial. To minimize these costs, yet keeping within strict bounds of safety, maximum possible use is made of nature's dilution capacity. Site selection for nuclear facilities plays a very important role in this regard.

The major operation of this kind is at Hanford and is related directly to the environmental situation existing there. Hanford Atomic Products Operation discharges approximately 50,000 curies/month and expects to continue discharging about the same amount in the future. Sr and Cs have been scavenged from this activity so that it essentially consists of short-lived material when discharged. Nitrate ion and ruthenium, and more recently cobalt 60, have been detected in monitoring wells associated with these operations but not as health-significant levels to date.

Some liquid wastes at the smaller sites are concentrated to relatively small volumes, packaged and then shipped to the larger sites for land burial or to the oceans for burial at sea. The following table, far from accurate and complete, provides a rough estimate of volumes of low-level liquid wastes:

Low-Level Liquid Waste Discharge Data

Site	Type of Maximum Treatment	Approximate Volume Discharged Per Year (10^6 gals)	Total Curies Discharged ($\times 10^3$)
ANL	Depends on specific waste	47	
BNL	Evaporation	120	(2-3)(1)
FMPC	Calcination	90	
HAPO	Decay Storage	7,668(3)	2,096 (since 1944)
KAPL	Evaporation	126	
LASL	Co-ppt	14	
NRTS	Decay Storage	420	1.7 (1955 and 1956)
ORNL	Soda-lime treatment	159	75 (since 1948)
Rocky Flats	Evaporation	40	
SRP	Decay Storage	20	0.8 (since 1955 and 1956)
WAPD	Evaporation	137(2)	-(1)
Total		8,841	2,173.5

*Activity disoharged without further control
(1) To sea burial in packages
(2) Includes infiltrated storm water
(3) Includes condenser cooling water not normally radioactive

Other More General Sources of Radioactive Wastes

The wastes described in the previous sections are all produced as more or less undesirable by-products of a major process; they are collected into large packages, reduced in volume, and stored with as little dispersal as possible. However, many of the radioactive isotopes, produced either from fission or by neutron capture, have use in small quantities as industrial and medical radioisotopes, and are sold in millicurie to curie quantities to thousands of users under liscenses issued by the AEC. The benefits of the radioisotope program have been great; the expansion of this program, number of users, and the use of much more energetic sources is foreseen. All these sources, unfortunately, must at some point, become waste materials.

Control of these widely distributed sources and their disposal has been established under the liscensing plan. The AEC Division of Inspection is responsible for maintaining good practice in the handling and disposal of the sources. Another control, more absolute in the sense that it can limit the total quantity of radioactive material "in circulation", is the one that can be exercised in limiting the quantities in use at the source of supply. It is an inherent feature of limitations already established by law that a general contamination problem affecting a large segment of the population cannot develop from the profitable use of radioactive sources and tracers.

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Representative HOLIFIELD. We have several interesting statements on "Nature of Wastes," which I would like to place in the record at this point.

(The statements referred to follow:)

NATURE OF WASTES FROM THE URANIUM MILLING INDUSTRY

Prepared by

Raw Materials Development Laboratory
Winchester, Mass.

INTRODUCTION

The recovery of uranium from its ores is presently performed in the United States entirely by hydrometallurgical processes. This is to say that the valued constituent of the ore is dissolved or leached with aqueous solutions. Following the leach, the uranium is recovered by a number of chemical reactions all of which contribute chemicals to the plant liquor. These liquors must eventually be discarded. The valuable material being processed, uranium, is the progenitor of a radioactive decay series which contributes radioactive isotopes to the solution in process. These isotopes must also be rejected from the plant. In view of the nature of the chemical process and the type of material being treated, the potential of chemical and radiochemical pollution exists. The degree of pollution and its effect on the environment surrounding any one plant is dependent on many variables.

TYPES AND LOCATION OF MILLS

The solutions used to dissolve the uranium may be either acidic with sulfuric acid or alkaline with sodium carbonate and sodium bicarbonate. The make-up of these leaching solutions provides a classification into which all uranium mills fall - acid or alkaline leach mills.

In the case of the acid process, the dissolved uranium may be selectively recovered and concentrated from the leach solution by the use of ion exchange resins or by the use of organic extractants. Acid leach mills may then be further classified by process as acid leach-ion exchange (IX) or acid leach - solvent extraction (SX). The uranium is recovered from the resin or solvent by stripping with an aqueous solution. A variety of chemicals may be used for this purpose, e.g., sulfuric acid and nitrate salts, sulfuric acid and sodium chloride, and sodium carbonate. The uranium in the strip solution is finally precipitated by the addition of caustic reagents such as sodium hydroxide, lime, or magnesia.

Following the leaching of the ore in the alkaline process, the leach liquor is clarified by filtration to remove the ore pulp, and sodium hydroxide is added to precipitate the uranium from solution. Thus the alkaline process may be further described as alkaline leach-filtration. This sodium hydroxide is later converted to sodium carbonate and sodium bicarbonate by the addition of carbon dioxide. Generally then, in the simpler alkaline flowsheets only sodium carbonate and bicarbonate are found in the waste solutions arising from chemicals added in processing the ore.

In both acid and alkaline processes, chemical oxidants may be added to accelerate the dissolving of the uranium by the oxidation of the uranium mineral. This step contributes manganese chloride and chlorate salts to the acid leach waste solutions and permanganate, manganese, copper and ammonia salts to the alkaline process tailings.

On the average, the acid plants will discharge about four tons of waste solution for each ton of dry ore treated. Alkaline plants will discharge much less - perhaps as little as one ton of solution per ton of ore. Essentially all of the solid material comprising the ore is also discharged to waste. These solids, with reasonable care, can be readily impounded and do not escape to the surrounding countryside.

Those portions of the ore that are soluble in the leaching solution being used also contribute to the make-up of the waste liquor. Usually no large amounts of chemicals are dissolved from the ore. However, certain radioactive daughter products of uranium are dissolved. These radioactive elements pose a serious problem of their own and will be discussed separately although from the engineering standpoint, chemical and radioactive pollution are interrelated.

In Table I are listed the uranium ore processing plants in the United States. The plants are classified according to leach process; and the operating company location, and rated capacity are tabulated for each mill.

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Table I

Uranium Ore Processing Plants

A. Acid Leach	Location	Capacity in Tons of Ore Per Day
Union Carbide Nuclear Company	Uravan, Colorado	1,000
Union Carbide Nuclear Company	Rifle, Colorado	400
Union Carbide Nuclear Company	Greenriver, Utah	300
Union Carbide Nuclear Company	Slick Rock, Colorado	300
Climax Uranium Company	Grand Junction, Colorado	330
Vanadium Corporation of America	Durango, Colorado	750
Trace Elements Corp. (UCN)	Maybell, Colorado	300
Vitro Uranium Company	Salt Lake City, Utah	600
Uranium Reduction Company	Moab, Utah	1,500
Texas-Zinc Minerals Corp.	Mexican Hat, Utah	1,000
Kerr-McGee Oil Industries, Inc.	Shiprock, New Mexico	300
Mines Development, Inc.	Edgemont, South Dakota	400
Rare Metals Corp. of America	Tuba City, Arizona	300
Western-Nuclear Corp.	Split Rock, Wyoming	400
Dawn Mining Company	Ford, Washington	400
Lucky Mc Uranium Corp.	Gas Hills, Wyoming	750
Gunnison Mining Company	Gunnison, Colorado	200
Kermac Nuclear Fuels Corp.	Grants, New Mexico	3,300
Lakeview Mining Company	Lakeview, Oregon	210
		12,740
B. Alkaline Leach		
U. S. Atomic Energy Commission	Monticello, Utah	350
Homestake-New Mexico Partners	Grants, New Mexico	750
Homestake-Sapin Partners	Grants, New Mexico	1,500
Phillips Petroleum Company	Grants, New Mexico	1,725
		4,325
C. Acid and Alkaline Leach		
The Anaconda Company	Bluewater, New Mexico	3,500
Fremont Minerals, Inc.	Riverton, Wyoming	500
		4,000
	Total	21,065

SOURCE AND CHARACTERISTICS OF CHEMICAL WASTES

The composition of wastes from uranium mills can be determined with fair accuracy by calculation of hypothetical effluent analyses by material balances around various mill circuits. Hypothetical balances were calculated for five variations of the two main processes: acid-resin in pulp (RIP), alkaline-filtration, mixed acid-RIP and alkaline-filtration, acid-countercurrent decantation (CCD)-column ion exchange, and acid CCD-solvent extraction (SX). The acid-CCD-SX circuit was investigated on three counts, calculated mill effluents being determined for a circuit using di-2-ethylhexyl phosphoric acid (EHPA) as the extractant with or without vanadium recovery, and a circuit using an unsaturated alkyl amine as the extractant.

These material balances were calculated using applicable ore and concentrate analyses, reagent consumptions in accordance with experience in the United States, typical leach liquor analyses; and for the case of waste solutions neutralized with lime to pH 7, accepted solubilities of the appropriate hydroxides in water. Results of the calculations are discussed in terms of the Utah Class "C" water specification because they are largely derived from drinking water standards suggested by the U. S. Public Health Service and are specific for the pollutants found in mill effluents. These specifications are presented in Table II. It should be noted that these specifications are for the receiving waters and are not restrictions on the composition of the plant effluent. The calculated mill effluent values were determined using as a basis 10,000 pounds of ore, generally assuming 90 per cent uranium recovery and 92 per cent uranium solubilization.

The hypothetical mill effluent analyses as determined by material balance are summarized in Tables III and IV, and these are compared with the appropriate analyses of actual mill tailings pond and stream samples. These actual samples were obtained to check the calculated analyses, and to investigate the presence of small amounts of those contaminants which may originate in the ore and for which specifications are stringent.

It should be noted that the theoretical balances cannot precisely predict the makeup of a mill effluent stream because of differences in the kind and quantity of reagents used, feed character, and operating conditions in mills of the same type.

Table II

STATE OF UTAH - CLASS "C" WATER SPECS

"No person may discharge into Class 'C' waters any waste which will result in chemical characteristics of said waters exceeding the limits prescribed by U. S. Public Health Service Drinking Water Standards." These standards are tabulated below

Impurity	Specification PPM by Weight
Lead	0.1
Fluoride	1.5
Arsenic	0.05
Selenium	0.05
Chromium - hexavalent	0.05
Copper	3.0
Iron and Manganese	0.3
Magnesium	125
Zinc	15
Chloride	250
Sulfate	250
Phonolic Compounds as Phenol	0.001
Total Solids	1000
Turbidity (Silica Scale)	10 PPM

Impurity	Other Specifications
Color	20 (platinum cobalt scale)
pH Range	5.0 - 9.0
Conductivity	< 2,250 micromhos/cm
Boron	1 PPM
Nitrate	20 PPM (?)
$SAR = \frac{Na^{+}}{\sqrt{\frac{Ca^{++} + Mg^{++}}{2}}}$	<30 in all cases - See Fig. 25 Agr. Handbook 60, U.S.D. Agr. Na, Ca, Mg in meqts. per liter.

Table III

Calculated Neutralized Tailings Pond Overflow Compositions Compared with Actual Values for Various Types of Uranium Mills

Waste Const. PPM	Acid RIP			Alkaline Filtration			Mixed Acid RIP-Alk. Tails		Acid CCD IX Columns Calc.	Acid CCD-SX EHPA No V Rec. Calc.	Acid CCD-SX EHPA With V Rec. Calc.	Acid-CCD Amine SX		Utah Class C Spec.
		Actual			Actual			Actual					Actual	
	Calc.	Mill A[1]	Mill B[2]	Calc.	Mill C[3]	Mill D[4]	Calc.	Mill C[3]				Calc.	Mill D[4]	
Cl	-	190				81			615	96	113	-	110	250
SO_4	4,556	3,860	2,210	3,870		1,760	4,700	4,490	3,194	3,706	4,006	4,560	2,910	250
Mg	430	535	42	1.1		< 10	370	325	200	56.6	56.6	275	72	125
NO_3	1,470	1,270	2,680	0		-	1,260	1,830	0	0	0	140	-	20
Cu		1.3	< 1.0	2.7		-	0.4	< 0.1	3.9	< 0.6	< 0.6	4	-	3.0
F	-	-	1.2			-		1.7	-	< 5.6	< 5.6	-	3.8	1.5
B	-	0.1	0.2			-		0.2	-				-	1.0
Fe	0.1	42	0.1	0.1	< 0.1	-	0.1	< 0.1	0.1	0.1	0.1	0.1	220	0.3
Mn	332	110	0	0		-	285	170	372	395	395	740	30	
Pb	-	0.65	0.04			-		0.19	-				-	0.1
As	-	0.21	< 0.01			-		0.06	-				-	0.05
Cr^{+6}	0.05			-		-	0.05		< 0.4	< 0.6	< 0.6	< 1.1	-	
U_3O_8	12.5	3.4	0.3	33.2	16	9.9	15.1	21	5.4	17.0	17.0	22.0	-	10
Na	250		910	3,610		3,450	685	1,230	405	665	808	28	-	
Ca	735	530	815	56.5		< 10	735	570	735	735	735	735	520	
HCO_3	0			2,560	1,590	1,100	328	410	0	0	0	0	0	
CO_3	-			-	8,210	4,610		-	0	0	0	0	0	
Mo	2.5			67		-	10.4		10.3			5.0	-	
NH_4	430			0		-	370		0	68	68	175	-	
Si	-	-		46		-	6		208	283	283	275	-	
V_2O_5	4.5			650		< 10	88		12.5	100	34	100	-	
Dissolved Solids	8,221	7,360	6,410	10,903	12,300	8,270	8,853	8,940	5,760	6,122	6,516	7,059	4,370	1,000
Suspended Solids		30	210	-	-	-								
SAR	1.8		8.4	131	-	-	5.1	10.2	3.4	6.4	7.8	0.2	-	30
pH	7.0	3.3	7.7		10.3	10.8	7.0	6.9				7.0	2.6	5.9
Organic	0			0			0			283	283	160	-	

(1) Composite samples taken November, 1957 (unneutralized).

(2) Composite samples taken September 8-14, 1957.

(3) Composite samples taken October 30 - November 6, 1957 (unneutralized).

(4) Twenty-four hour sample taken March, 1958 (unneutralized).

Table IV

Effect of Tailings Pond Discharge on the Chemical Composition of a Large River at Mill A and on a Small Stream at Mill C

Combined Acid RIP - Carbonate Mill

Waste Constituent PPM	C_2 Tailings Pond Discharge Mill C(1)	C_1 Stream Above Mill Mill C(1)	C_3 Stream Below Mill Mill C(1)
Cl	-		
SO_4	4,490	82	2,970
Mg	325	12	130
NO_3	1,830	120	1,040
F	1.7	0.6	1.5
B	0.2		
Fe	<0.1	<0.1	<0.1
Mn	170	<0.1	110
Pb	0.19		
As	0.06		
U_3O_8	21	<0.1	15
Ca	570	57	460
Dis. Solids	8,940	430	5,760
Susp. Solids	20	120	1,920
pH	6.9	7.8	7.3

Acid RIP Mill

Waste Constitutent PPM	C_2 Tailings Pond Discharge Mill A(2)	C_1 River Above Mill Mill A(2)	C_3 River Below Mill Mill A)2)
Cl	190	58	140
SO_4	3,860	480	600
Mg	535	48	48
NO_3	1,270	26	31
F	-	0.9	0.9
B	0.1	0.5	0.2
Fe	42	<0.1	<0.1
Mn	110	<0.1	<0.1
Pb	0.65	0.18	0.17
As	0.21	0.04	0.04
U_3O_8	3.4	<0.1	<0.1
Ca	530	115	135
Dis. Solids	7,360	1,120	1,170
Susp. Solids	30	80	180
pH	3.3	7.6	7.5

(1) Samples taken October 30 to November 6, 1957
(2) Samples taken November, 1957

These hypothetical tailings pond effluent analyses are, however, useful in defining the order of magnitude of the various pollutants which are likely to appear in the tailings pond effluent stream from each type of process.

The results of these calculations and determinations of the nature and extent of chemical stream pollution suggest the following conclusions:

1. Manganese on the basis of laboratory testing will be the most serious offender when MnO_2 oxidant is used since the specification for the sum of iron and manganese is very low. Five pounds of MnO_2 per ton of ore would result in a soluble manganese concentration of 200 to 400 PPM in tailings neutralized to pH 7. The iron and manganese specification would then be exceeded about 1000-fold. The stream flow (dilution) required for specification waste disposal would on this basis be unreasonable.

2. Nitrate can be expected to pose a serious problem in those mills utilizing nitrate elution of resin. Nitrate pollution will be important but somewhat less serious in mills employing nitrate stripping of a solvent.

3. Sulfate and total dissolved solids are major offenders in all mills.

4. Magnesium will exceed specification in acid leach mills and chloride in mills using chloride elution. Copper and chromium may exceed specification.

5. There are a large number of possible offensive materials which can neither be estimated nor calculated, as they originate mainly from the ore. These include fluoride, boron, selenium, lead, arsenic, and phenolic compounds. As specifications for these materials are stringent, direct sampling of tailings ponds will be necessary to establish the level of these substances and any other possible offender.

The data presented in Tables III and IV further indicate:

(a) The tailings pond at Mill B at pH 7.7 is over Utah Class "C" specifications only with respect to nitrate, sulfate, and total dissolved solids. The other constituents investigated were all below the specified limit. As no oxidant is used in the circuit and tails are sufficiently neutralized to precipitate ferrous hydroxide, the iron and manganese specification

was met. The Mill B tailings pond was not regularly discharging into the neighboring river because of restrictions on radioactive pollution and therefore no river water samples were studied.

(b) The Mill C tailings pond overflow at pH 6.9 exceeded specification with respect to iron and manganese, nitrate, sulfate, total dissolved solids and, to a minor extent, magnesium. The uranium content of 0.021 gram U_3O_8 per liter was also excessive. Other constituents met specifications or exceeded them by only a small factor. Analyses of creek water samples above and below the mill indicate serious contamination by iron and manganese, nitrate, sulfate, and dissolved solids. The uranium content of the creek below the mill was 0.015 gram U_3O_8 per liter. It should also be noted that the nitrate content of the creek above the mill was even then well over specification, possibly due to wastes from a nearby town.

The average dilution of pond overflow available at Mill C at the time of sampling (October, 1957) was only about 50 per cent, with the pond overflow rate about twice the flow of the creek. It would seem necessary at these dilution levels to attempt to meet waste specifications at the tailings pond without regard to the small amount of dilution available.

(c) The tailings pond overflow at Mill A exceeded specification with respect to iron and manganese, nitrate, sulfate, dissolved solids, lead, arsenic, and magnesium. The acidity was also excessive as the pond overflow discharged at pH 3.3 into the river. The mill apparently does not cause serious contamination of the river, although sulfate, nitrate, and dissolved solids concentrations already slightly above specification in the river were somewhat increased.

The magnitude of the dilution available at Mill A allows this facility to discharge solution well over specifications in some respects without seriously affecting the level of contamination in the river.

SOURCE AND CHARACTERISTICS OF RADIOACTIVE WASTES

This discussion has been limited to a consideration of the concentration of chemical contaminants in waste solutions from uranium mills. The presence of radioactive material in these tailings and the resultant pollution problem should also be considered.

Radioactive substances are present in uranium ores because the natural uranium isotopes U^{238} and U^{235} are the parents of the "uranium" and "actinium" series of naturally occurring radioactive elements. The natural abundance of these two parent isotopes is 99.28 per cent for U^{238} and 0.715 per cent for U^{235}. The actinium series contributes very little to the total activity of uranium deposits, and it can be ignored.

Uranium-238 decays by several alpha-particle and beta-particle emissions until it reaches the stable end product lead-206. If the uranium ore is old enough, the daughter elements are in secular equilibrium with the uranium parents; i.e., they are being formed as fast as they are decaying so that their activity is the same as the parent (7.5×10^5 dpm per gram of U^{238}). Once the equilibrium is disrupted, e.g., by leaching or removal of the uranium from solution by ion exchange or solvent extraction, many of the daughter elements will disappear because of their short half-lives. However, there are many isotopes of long half-lives which will survive and constitute a potential radioactive pollutant in mill waste streams. The permissible concentrations of radioactive isotopes in water discharged to unrestricted areas has been defined by AEC regulations as set forth in Part 20 of the Federal Register, Vol. 22, No. 19 (Jan. 22, 1957). The permissible levels for those elements which may be found in uranium mill waste streams, along with their half-lives and modes of emission are listed in Table V on the following page.

Table V

List of Potential Radioactive Contaminants in Uranium Mill Wastes

Isotope	Half Life	Radiation	Permissible Concentration Unrestricted Area 1/	
			μc/ml	dpm/l
U-natural		alpha	7×10^{-6}	1.5×10^{4}
Th^{234}	24.1 days	beta	3×10^{-1}	6.6×10^{8}
Th^{230} 2/	8.0×10^{4} years	alpha		
Ra^{226}	1622 years	alpha	4×10^{-9}	8.8
Rn^{222}	3.825 days	alpha	2×10^{-7}	4.4×10^{2}
Po^{210}	138.4 days	alpha	3×10^{-6}	6.6×10^{3}

1/ In this discussion radiochemical concentrations are presented in units of disintegrations per minute per liter, dpm/l, and microcuries per milliliter, μc/ml. The dpm/l units are convenient for expressing experimental data, but the Part 20 specifications are given in units of μc/ml. To convert dpm/l to μc/ml divide by 2.22 x 109, e.g., the specification for Ra^{226} is 8.8 dpm/l or 4.0×10^{-9} μc/ml.

2/ Not presently defined but is expected to be stated in the forthcoming new issue of Part 20.

In order to determine which radioactive isotopes were present in mill effluents in excess of permissible concentrations, samples of clarified tailing pond water from the various types of uranium mills were analyzed. These analyses are listed in Table VI, from which it can be seen that the uranium concentration was too high in 2 out of 6 effluents, the radium specification was exceeded in all cases, the thorium -230 concentration was probably too high in 2 out of 5 cases, and the thorium -234 specification was not exceeded at all.

Table VI

Analyses of Tailings Pond Effluents

	pH	U_3O_8 g/l	Tailings Effluent Analyses					
			Ra 226		Th 230		Th 234 + Pa 234	
			µc/ml	dpm/l	µc/ml	dpm/l	µc/ml	dpm/l
1.	1.5	0.10*	4.0×10^{-6}*	8,900	3.9×10^{-5}	85,000	1.2×10^{-4}	270,000
2.	3.3	0.0034	5.4×10^{-6}*	11,800	1.2×10^{-6}	2,600	4.8×10^{-6}	10,500
3.	7.7	0.0003	3.3×10^{-7}*	716	2.8×10^{-8}	61	1.8×10^{-7}	404
4.	6.9	0.02*	8.1×10^{-8}*	178	9.4×10^{-8}	206	5.1×10^{-6}	11,320
5.	9.9	0.009	2.0×10^{-6}*	4410				
6.	2.0	0.002	2.7×10^{-6}*	5960	1.1×10^{-5}	24,700		

Mill Processes:
1. Acid Leach - Solvent Extraction
2. Acid Leach - RIP
3. Acid Leach - RIP
4. Combined Acid Leach - RIP and Alkaline Leach - Precipitation
5. Alkaline Leach
6. Acid Leach - CCD - Solvent Extraction

* Specification exceeded

SUMMARY

In summary, the liquid wastes generated by the uranium milling industry have been determined to be sources of chemical and radioactive pollutants. The chemical species may include iron and manganese, sulfate, nitrate, magnesium, uranium, chloride, and dissolved solids. The radioactive isotopes in uranium mill tailings include natural uranium, thorium -234, thorium -230, and radium -226. The specific pollutants depend upon the particular flowsheet considered. The severity of the pollution problem depends upon the size, quality, and ultimate usage of the receiving waters, while the legal aspects depend upon the specific state laws involved.

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THE NATURE OF WASTES PRODUCED IN THE MINING AND MILLING OF URANIUM ORES

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The mining and processing of uranium ores involves the handling of large amounts of unwanted materials to obtain a small amount of the desired product, uranium concentrates. Domestic ores average about 0.3 percent U_3O_8, or about six pounds per ton of ore. All of the rest of the ore that enters the mills is waste which must be separated and discarded, an exception being that in certain mills vanadium is recovered. Additionally, in the mines, varying proportions of waste material must be handled in order to obtain a ton of ore. This mine waste will vary in composition from barren rock to material containing almost enough uranium to be worth shipment. Thus, as in all mining operations, the definition of waste rock is flexible and will depend on several factors, such as the size of the ore bodies, mining methods required, and cost of transportation to market. In general, to a miner, anything that is not shipped to a mill is waste, and to a mill operator, anything received that is not barrelled and sold is waste.

Uranium ores contain all the members of the uranium 238 and uranium 235 series, a total of 26 radioactive isotopes. In addition, the ores may contain other compounds such as free silica, vanadium, arsenic, or selenium, which are known to be toxic. All of these substances, except the uranium, are waste and their presence must be

considered in the mining and milling operations. In some cases, the non-radioactive wastes will present the most troublesome handling and control problems, while in others primary attention must be given to the radioactive materials.

The quantities of radioactive wastes produced in uranium mining and milling cannot be determined positively, but estimates can be made. If the ores were in equilibrium, each member of the uranium family would be present in equal quantities, curiewise. Analyses of many thousands of samples by the U. S. Bureau of Mines have shown that the ores are not in equilibrium, probably due to selective leaching of the mineral bodies. However, approximate amounts can be calculated by assuming that equilibrium exists.

The present rate of production of domestic uranium is about 65,000 pounds per day, or 9.8 curies of uranium 238. At equilibrium, this would mean that 9.8 curies of each of the other members of the uranium series would be associated with this quantity of uranium 238. Uranium 235 and its daughters would also be present in much smaller amounts (about one part in 23 on an activity basis). From these assumptions, it seems that in the order of 100 curies of solid waste radioactive elements are delivered to the mills each day.

In the mines, the most important radioactive waste products which reach the outside air are the only gaseous member of the uranium series, radon, and its short-lived daughters. As radon is a gas, it diffuses into open underground areas from the surrounding materials. Non-commercial uranium ores, or the host rock, are sources of radon

as radium (the parent of radon) is widely distributed in nature. The quantity of radon emanated will depend on the amount of radium in the surrounding rock, the porosity of the rock, and other factors. Studies made in a number of mines have measured emanation rates ranging from 9×10^9 to 2×10^{11} atoms of radon per 1000 cubic feet of mine volume, or from 5×10^{-7} to 2×10^{-5} curies per minute per 1000 cubic feet. Radon has a half life of 3.8 days, which is long compared with the time needed to change the air in a mine, and so most of the gas emanated into a mine can be considered to be removed eventually. It is impossible to estimate the amount of radon that is dispersed into the atmosphere by uranium mines. However, the total quantity is very small compared to that which is naturally produced from the surface of the earth.

Extensive studies have shown that inside the mines the most important radioactive contaminants of the atmosphere are radon and its daughter products. Detailed reports[1][2][3][4] of these investigations have been published and should be studied for complete discussions of the problems encountered and the control measures needed. The following tables from the Report on Radon Daughter Product Samples, 1957,[4] summarize the most recent data obtained on the mines. These samples were taken by state health departments, state bureaus of mines, mining companies, and the staff of the Occupational Health Field Station.

Table 1 shows the coverage obtained in 1957. In Arizona and New Mexico samples were taken in almost all of the operating mines

and probably are representative of the situation in those states. In Colorado and Utah the coverage was far from complete and represents only the mines in which surveys were made.

TABLE 1

NUMBER OF MINES REPORTING AND NUMBER OF SAMPLES BY STATE, COMPARED WITH NUMBER OF MINES FOUND IN THE1957 SUMMER CENSUS

State	No. Mines Found	No. Mines Reporting	No. Samples
Arizona	35	26	68
Colorado	213	81	233
New Mexico	30	32	241
Oregon	1	1	7
Utah	88	29	375
	367	169	924

Table 2 lists the frequency distribution of radon daughter concentrations by states. The summary indicates that in Utah and New Mexico many of the reporting mines were using adequate control methods, while in Arizona and Colorado many miners were exposed to high concentrations of radon daughters.

TABLE 2

FREQUENCY DISTRIBUTION OF RADON DAUGHTER CONCENTRATIONS BY STATE, 1957

Concentration* (x WL)	Arizona	Colorado	New Mexico	Oregon	Utah	Total
< 1	18	40	109	1	141	309
1 - 1.9	12	25	38	-	85	160
2 - 4.9	6	51	32	3	82	174
5 - 9.9	10	36	18	1	35	100
10 - 19.9	13	25	21	2	20	81
20 - 50	5	20	18	-	11	54
> 50	4	36	5	-	1	46
Total	68	233	241	7	375	924
Range of Conc.	0-162	0-154	0-133	0.8-11.3	0-66.4	0-162
Median Conc.	4.1	5.1	1.3	4.8	1.5	1.9

*WL = Working level or 1.3×10^5 Mev of potential alpha energy per liter of air.

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All the other constituents of the ore and waste rock will also be present in the dust produced by mining operations. Probably the most toxic of the radioactive elements is radium, and so its occurrence in mine atmospheres has been studied. About 400 atmospheric samples have been analyzed, of which about 10 percent showed atmospheric concentrations greater than the threshold level for this element. These results indicate that under present operating conditions air-borne radium is not a significant health problem in uranium mines.

Important non-radioactive wastes in mine air would include such materials as free silica dust, diesel engine exhausts, and powder gases. The toxicity of these materials is well known as they are encountered in many industries. Standard procedures which were developed for other operations must be employed in uranium mines to control such hazardous compounds. These would include wet drilling, wetting the muck piles and haulageways, and providing sufficient ventilation to reduce the atmospheric concentrations of dust and gases to or below the threshold limits. In many mines the amounts of fresh air required to control exposures to these compounds will also be sufficient to control the exposure to radioactive materials.

Waste materials in uranium mills present problems which differ in kind and degree from those encountered in mines. In the sampling, grinding, and crushing areas, dust is produced containing all the constituents of the ore. Based on data from detailed studies [5][6] of almost all of the operating mills, it is indicated that the primary health hazard in these areas is produced by silica dust. The following tables taken from the Review of Environmental Studies in Uranium Mills (1950-1956)[5] summarize

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TABLE 3

FREQUENCY DISTRIBUTION OF DUST COUNTS - COMPOSITE OF ALL URANIUM MILES SURVEYED

Concentration MPPCF*	Crusher Areas	All Mill Areas
0 - 5	16	88
5 - 10	21	60
10 - 20	30	77
20 - 50	22	66
50 - 100	13	36
>100	22	35
Total	124	362
Range	1 - 1,250	<1 - 1,250
Median	18.9	14.4

*Million particles per cubic foot of air.

TABLE 4

FREQUENCY DISTRIBUTION OF AIR-BORNE URANIUM CONCENTRATIONS - COMPOSITE OF ALL URANIUM MILLS SURVEYED

Concentration T.L.*	Concentration ug/M^3**	Crusher Areas	Final Product Areas	All Mill Areas
< T.L.	< 250	86	51	375
1- 2xT.L.	250- 500	9	4	33
2- 5xT.L.	500-1,250	8	6	24
5-10xT.L.	1,250-2,500	1	3	9
10-20xT.L.	2,500-5,000	7	1	9
>20xT.L.	>5,000	1	18	19
Total		112	83	469
Range		1 - 5,100	2 - 245,000	1 - 245,000
Median		164	211	157

* Threshold limit.

** Micrograms per cubic meter.

FREQUENCY DISTRIBUTION OF AIR-BORNE RADIUM CONCENTRATIONS - COMPOSITE OF ALL URANIUM MILLS SURVEYED

Concentration T.L.*	Concentration uug/M^3**	Crusher Areas***	Final Product Areas#	All Mill Areas&
< T.L.	< 24	17	3	89
1- 2xT.L.	24- 48	7		16
2- 5xT.L.	48-120	6	2	14
5-10xT.L.	120-240	1		3
10-20xT.L.	240-480	1		1
>20xT.L.	> 480	1	—	2
Total		33	5	125
Range		2 - 970	0.7 - 60	0.2 - 970
Median		24.0	--	17.0

* Threshold limit.

** Micrograms per cubic meter of air.

*** Results from 5 plants only.

Results from 2 plants only.

& Results from 6 plants only.

TABLE 6

FREQUENCY DISTRIBUTION OF AIR-BORNE VANADIUM CONCENTRATIONS - COMPOSITE OF ALL URANIUM MILLS SURVEYED

Concentration T.L.*	Concentration ug/M^3**	Crusher Areas***	Final Product Areas#	All Mill Areas***
< T.L.	<500	53	42	239
1- 2xT.L.	500- 1,000	18	1	35
2- 5xT.L.	1,000- 2,500	9	5	34
5-10xT.L.	2,500- 5,000	1	0	9
10-20xT.L.	5,000-10,000	2	0	4
> 20xT.L.	>10,000	1	1	3
Total		84	49	324
Range		10 - 20,000	5 - 20,114	0 - 20,114
Median		400	250	340

* Threshold limit.

** Micrograms per cubic meter of air.

*** Results from 9 plants only.

Results from 5 plants only.

the data given in the report. Table 3 shows that in the ore crushing areas 14 percent of the samples were at or below the threshold limit of 5 million particles of dust per cubic foot of air, while 86 percent were above this limit. Table 4 shows that in the crusher areas 56 percent of the samples showed atmospheric concentrations of uranium at or below the threshold limit. Table 5, which gives the data on radium, shows that 52 percent of the samples in the crusher areas were at or below the threshold limit for radium. As the most toxic radioactive element in the ores is probably radium, a study was made of a limited number of crusher operators to evaluate their absorption of this element. None of the workers had built up a detectable body burden of radium. These results indicate that air-borne radium is not an important problem.

In the chemical processing sections of the mills, the waste products are mainly gases and vapors from chemicals added to the slurries and solutions, or produced by the action of these chemicals on the ores. Highly toxic compounds such as hydrogen sulfide, arsine, or acid gases can be generated and must be controlled. No estimate can be given of the amounts of such wastes discharged to the outside air.

The final processing area handles uranium separated from all but traces of other radioactive elements. Here, the primary wastes are uranium compounds which are disseminated into the workroom air. Exceedingly good control methods and dust collectors are required to keep the atmospheric concentrations of uranium below the threshold level and to prevent loss of the product. For example, in one mill, stack samples indicated that approximately one ton of uranium a year was passing through the collector.

It was mentioned previously that practically all of the ore received by the mills is discarded as waste slimes and sands. In addition, the chemicals added during the processing are also discharged to the tailings pond or in other plant effluents. These chemical wastes can be important sources of water pollution and their release to the environment must be controlled in accordance with standard practices. Additionally, 12,240,000 gallons of water are used each day, most of which is discharged as plant effluents carrying the dissolved and solid wastes to the tailings ponds.

The most important radioactive element in the wastes is radium which is practically all discharged in mill effluents. Thus, approximately 9.8 curies a day of this substance appears as waste, in a concentration of about one milligram per ton of solids. Several preliminary studies have been made to measure the concentration of radium in water courses and to determine the effects of this element on the receiving waters. A report of these studies [7] is discussed in detail in other papers submitted for the record. The available data indicate that of the radium which enters the mills a large proportion goes first to the tailings ponds and the remainder is discharged to water courses as dissolved radium or as part of the suspended solids in the pond effluents.

At this time sufficient information is not available to permit definite conclusions concerning the ultimate effect on the environment of the toxic and radioactive wastes released by uranium mills. Data are needed on such factors as the rate of removal of these elements

from the tailings ponds by weathering and their ultimate disposition, such as absorption by plants, fish, animals, and downstream populations.

In summary, uranium mining and processing involves the handling and disposal of a large number of waste products. These materials are composed of solids, liquids, and gases, some of which are common to many industries, and some radioactive elements which are peculiar to this one. In mining, the most significant materials of hygienic importance are radon and its daughters and silica dust. These elements are potentially hazardous to the health of the miners, and exposure to them must be controlled. A variety of toxic materials are handled or produced in milling operations, which require control to prevent health damage to the workers. The effluents from the mills contain chemical and radioactive wastes which can present water and air pollution problems.

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URANIUM MILL WASTES

By

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The public is interested in uranium because it is a basic raw material in the rapidly developing atomic energy industry. Radioactive elements have brought man both pain and comforts; they will undoubtedly continue to do just this. Consequently, we must learn how to live with them safely.

Uranium oxide (U_3O_8) being a basic building block in the atomic energy industry is the prize sought by our miners and ore processors. Its production, while not overly hazardous, does introduce some complex problems of far reaching consequence. It is a radioactive element, and it is the jumping off place for several other such elements.

We have an interest in radioactive elements because of the way they can affect our environment. The uranium-radium series offers understanding of the radium and other radioactive elements present in uranium mill wastes. We are definitely interested in radium because the radiation it produces. (See Table 1 and Figure 1) Radium^{226} emits alpha radiation, which is a bone seeker and has a 1,612 year half life. Spreading radium bearing waste material indiscriminately is inviting obvious trouble. Values have been established for the maximum permissible amount of this type radiation to which man may be exposed. The subcommittee on Permissible Internal Dose of the National Committee on Radiation Protection recommends that the maximum permissible concentration of radium^{226} + its daughter elements (both in equilibrium) for man to be 4×10^{-8} micro curies per milliliter of water; this being for continuous exposure.

Of course, the Committee states that no radiation at all is the ideal exposure to work for and certainly such is desirable. However, working with radioactive materials requires that the worker be subject to some exposure and a certain amount of radiation can be tolerated.

The importance of radium to water works operators is that as part of the discarded mill wastes some of the radium dissolves in the stream water. Radium is an element of the carnotite ore just as is uranium, but unlike uranium, radium now has no market value. Rather it is discarded just as are the tailings, sands and slimes of the original ore. This being the case, it is important that we examine this problem for its size, for its effect upon the environment, and for possible solutions leading to the prevention of any possible hazards resulting from water contamination by mill wastes.

As to the size of the problem resulting from the wastes incident to the processing of uranium ores it is confined to a relatively small area of the United States. The principal ore producing area is on the Colorado plateau comprising portions of Colorado, New Mexico, Arizona and Utah with some additional production in South Dakota, Wyoming and Washington. Uranium ore production in the United States totaled 2½ million dry tons for the first six months of 1958 as compared to approximately 2 million dry tons for the last six months of 1957, an increase of ½ million tons. Estimated reserves of uranium bearing ores as of June approximately 78½ million tons, - ores containing 0.28% uranium oxide (U_3O_8). This, in addition to 2 million tons of the same quality ore in stockpiles. Considering that the daily processing of 10,000 tons of this grade ore could produce approximately 2,500 grams of radium^{226} per year, which is 2½ times the estimated total supply of purified radium for all medical and industrial uses in the United States, the problem of mill wastes and how they are discarded becomes significant.

Uranium ore production is the province mainly of the miner and mine operator. Except for unusual finds of rich ore that present problems of too much radiation exposure (most ores assays at $\frac{1}{4}$% U^{308}) there appears to be no more than usual mining hazards.

Uranium ore processing consists essentially of separating a very small amount of uranium oxide from a very large amount of rock or earth. Consider separating 5 lbs. of uranium oxide from 2,000 lbs. of rock and dirt. The ore is crushed, ground, samples and then passes thru a variety of chemical processes eventually coming to rest as tailings, sands, and slimes on the one hand and a very small amount of marketable product on the other. The many processes are being improved upon each year. There is hope that economical methods can be developed that will bind chemically such products as radium to the product and the tailings so as to keep it from dissolving out in the receiving waters. At one mill neutralization of the slimes tailings with lime resulted in about 85% of the dissolved radium being precipitated. Consequently, it became a part of the yellow cake which is the substance the mill is in business to recover and sell; also it eliminates radium from the wastes.

There is importance in the fact that certain mill wastes may contaminate waters that are sources of supply to municipal and industrial water users. Should such wastes contain dissolvable radium they become very important because the maximum permissible concentration of soluble radium in domestic water is 40 micro micro grams per liter of water and it is recommended that this be reduced to one tenth that amount. Luckily, most of the radium found in the wastes from the uranium mills is in insoluble form and remains with the sands and slimes. Interestingly enough it has been estimated in the case of one mill that

about one percent of all radium wastes is soluble and this one percent must be given consideration. It has been shown that mill effluent results in some seepage from tailings ponds and thereby increases the amount of soluble radium in surface waters. River water collected in 1950 above and below four mills located on three rivers showed soluble radium to range from 22 to 130 times greater below the mill than above it. (See Table 2). Likewise data collected during 1955 showed the surface waters of the same river (See Table 3) upstream of the mill to have a soluble radium content ranging from 0.2 to 0.4 micro micro curies per liter, but downstream ranging from 0.3 to 86 micro micro curies per liter. Thus, the contribution of soluble radium to the streams by the mills is indicated to be as much as 200 times background. The soluble radium content of the stream biota ranged from 70 to 660 micro micro curies per gram (ashed weight) at downstream locations. Compared to uncontaminated stream biota, the increase was in the neighborhood of 100 times.

It is evident from even a brief discussion of presently available data that uranium mill wastes cannot be dismissed when they might contaminate surface waters. Admittedly, there is need for further study and this is going forward. The problem is one of searching for a solution, a method for reducing, or better still completely removing those radioactive elements from the mill wastes prior to discharge to surface or underground sources of water. This knotty task confronts the sanitary engineer, the mill operators, as well as the Atomic Energy Commission and the entire water drinking public in areas of uranium ore milling.

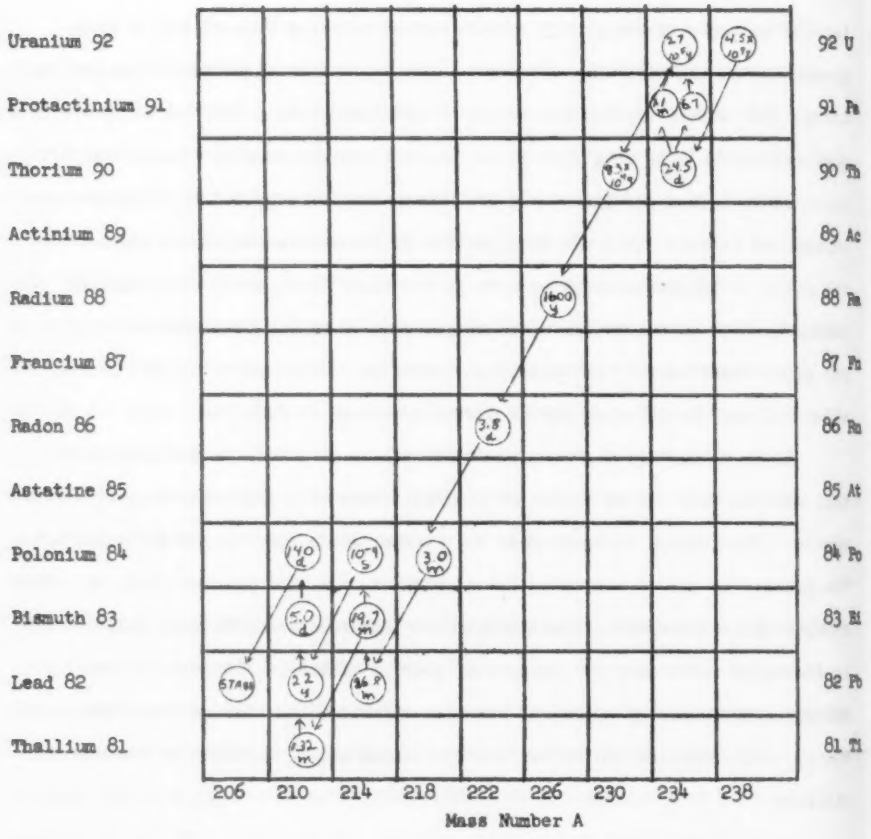

Fig. 1. The uranium series. Alpha decays are indicated by diagonal lines and beta transitions are shown as short vertical lines.

TABLE I.

THE URANIUM SERIES

ELEMENT	SYMBOLS		HALF LIFE	Energy of radiation (Mev)		
				Alpha	Beta	Gamma
Uranium	$92U^{238}$	(UI)	4.50×10^9 yr	4.21	-	-
Thorium	$90Th^{234}$	(UX_1)	24.1 days	-	0.13	0.09
Protactinium	$91Pa^{234}$	(UX_2)	1.14 min	-	2.32	0.80
Uranium	$92U^{234}$	(UII)	2.35×10^5 yr	4.76	-	-
Thorium (Ionium)	$90Th^{230}$	(Io)	8.0×10^4 yr	4.66	-	2
Radium	$88Ra^{226}$		1612 yr	4.79	-	0.19
Radon	$86Rn^{222}$	(Em)	3.825 days	5.49	-	-
Polonium	$84Po^{218}$	(RaA)	3.05 min	5.99	B	-
Lead	$82Pb^{214}$	(RaB)	26.8 min	-	0.65	0.29
Bismuth	$83Bi^{214}$	(RaC)	19.7 min	5.50	3.15	1.8
Polonium	$84Po^{214}$	(RaC')	1.5×10^{-4} sec	7.68	-	-
Thallium	$81Ti^{210}$	(RaC'')	1.32 min	-	1.80	-
Lead	$82Pb^{210}$	(RaD)	22.2 yr	-	0.026	0.047
Bismuth	$83Bi^{210}$	(RaE)	4.97 days	-	1.17	-
Polonium	$84Po^{210}$	(RaF)	138 days	5.30	-	2
Lead	$82Pb^{205}$	(RaG)	Stable	-	-	-

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TABLE 2

Water Samples of October 1950

Location	River	Flow, cfs.	Soluble Radium, uuc/l. Above Mill	Below Mill
Durango	Animas	300	0.20	4.5
Naturita	San Miguel	64	0.13	16.9
Uravan	San Miguel	64	4.4	43.4
Rifle	Colorado	1600	0.36	4.5

TABLE 3

Radioactivity of River Waters

River	Location	Radioactivity, uuc/l. Gross* Alpha	Gross* Beta	Soluble Radium	Soluble* Uranium
Animas	Above Durango	2	32	0.2	0.3
Animas	Below Durango	34	26	3.3	8.0
Colorado	Above Rifle	0	7	0.4	n.d.
Colorado	Below Rifle	49	15	0.4	n.d.
Colorado	Above Gr. Junction	13	22	0.3	n.d.
Colorado	Below Gr. Junction	183	295	39	n.d.
San Miguel	Above Naturita	10	16	0.3	2.8
San Miguel	Below Naturita	148	61	9.4	24
San Miguel	Above Uravan	92	46	4.9	21
San Miguel	Below Uravan(1)	4,700	5,490	86	n.d.
San Juan	Above Ship Rock(2)	220	369	0.2	n.d.
San Juan	Below Ship Rock (2)	170	204	0.3	n.d.
Jordan	Above Mill	5	28	0.4	n.d.
Jordan	Below Mill	23	51	n.d.	n.d.
Creek	Above Monticello	33	67	0.3	n.d.

* Total of suspended and dissolved fractions.

** Minima.

(1) Alpha activity above 50 per cent dissolved; beta activity above 12 per cent dissolved.

(2) Suspended solids very high (3150 and 4400 ppm); most of gross alpha and beta activity in suspended form.

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WASTE CHARACTERISTICS FOR THE RESIN-IN-PULP URANIUM EXTRACTION PROCESS*

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INTRODUCTION AND BACKGROUND

During the summer of 1957, the Public Health Service carried out a field study of liquid wastes resulting from the extraction of uranium from its ore in a typical refinery using the resin-in-pulp extraction process. The study was performed in cooperation with the South Dakota State Department of Health, the Mines Development Company of Edgemont, South Dakota, and the United States Atomic Energy Commission. It was the first of a series of such surveys planned by the Public Health Service in order to develop detailed knowledge of the characteristics of wastes, particularly radioactive, arising from the extraction of uranium from its ores. Specific objectives of the studies include detailed analysis of the extraction process, characterization of the resulting liquid wastes, evaluation of their water pollution and public health significance, and development of adequate and suitable waste control measures. At the same time, parallel field studies of the fate of these wastes in the water environment are being carried out.(1)

The uranium refinery at Edgemont, South Dakota, is located on the banks of the Cheyenne River about 35 river miles above Angustora Reservoir, a recreational lake (see Figure 1). This refinery is a typical example of the acid leach-resin-in-pulp process, of intermediate ore processing capacity. At the time of the field survey it was processing slightly more than 500 tons per day of ore that

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* A contribution from the Water Program, Robert A. Taft Sanitary Engineering Center, Public Health Service, Department of Health, Education, and Welfare, Cincinnati, Ohio.

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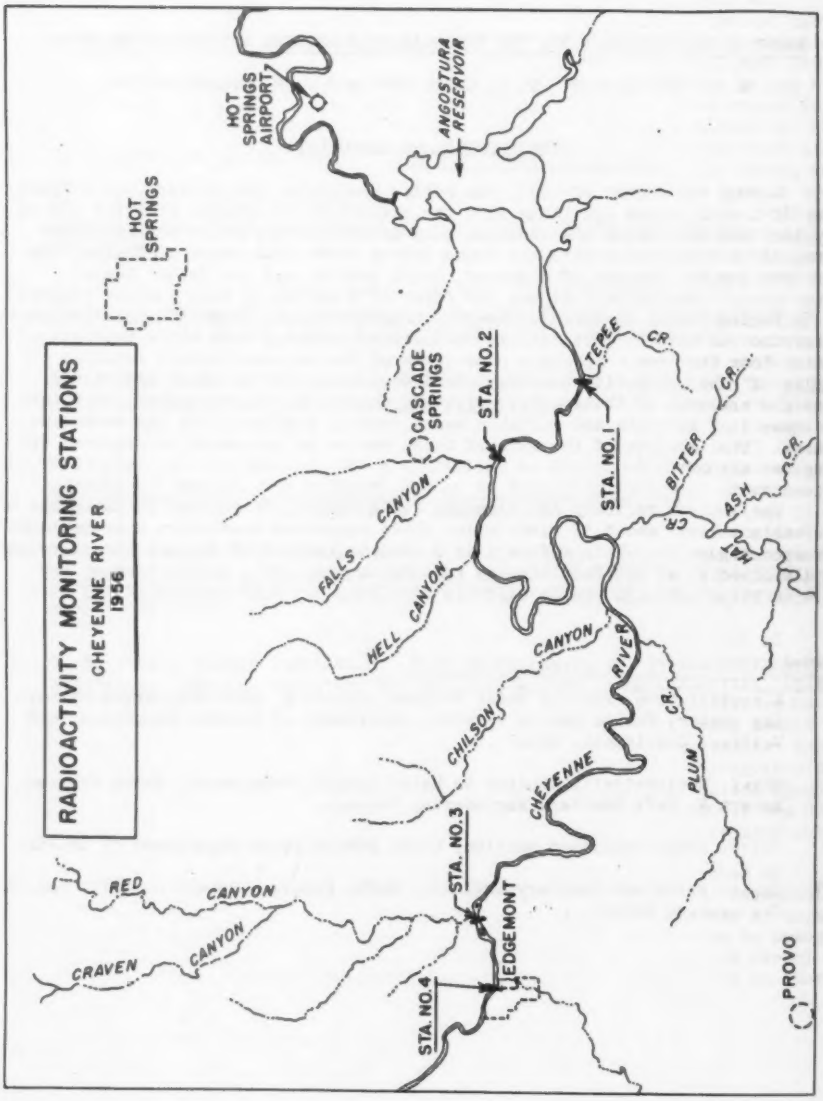
RADIOACTIVITY MONITORING STATIONS
CHEYENNE RIVER
1956
HOT SPRINGS
HOT SPRINGS AIRPORT
ANGOSTURA RESERVOIR
CASCADE SPRINGS
STA. NO. 2
STA. NO. 1
TEPEE CR.
BITTER CR.
ASH CR.
HAT CR.
FALLS CANYON
HELL CANYON
CANYON
CHILSON
CHEYENNE RIVER
PLUM CR.
STA. NO. 3
RED CANYON
CRAVEN CANYON
EDGEMONT
STA. NO. 4
PROVO

FIGURE 1

assayed about 0.20 per cent U_3O_8. The mill had been in operation about one year prior to this study. Virtually all liquid wastes were delivered to tailings ponds for storage and volume reduction by evaporation and seepage. There was a small direct liquid discharge to Cottonwood Creek, a tributary of the Cheyenne.

The Cheyenne River near Edgemont, South Dakota, is a relatively shallow stream having a sandy bottom. Flow is often turbid, and at the time of the field survey biological life was relatively sparse. River flows have been recorded from 1928-1933 and from 1947 to the present by the U. S. Geological Survey at a gaging station located just upstream from Cottonwood Creek. There are some 7,143 square miles of drainage area above this gage. During the period 1947 to 1955 the average annual flow in the river at Edgemont ranged from 20.5 to 144 cubic feet per second (c.f.s.); the minimum monthly average flow for this nine year period ranged as low as 0.02 c.f.s. and as high as 1.6 c.f.s. The flow drops to zero every year for varying periods - in 1952 there were only 6 days of zero flow, while in 1950 the flow in the river was zero for a total of 62 days. Over the nine year period there were a total of 296 days of zero flow - in other terms, the records indicate that on the average the river flow at Edgemont has been zero for 8.1 per cent of the days.

During February 1956, before the mill went into operation, a radioactivity background survey of the Cheyenne River below Edgemont was made by the South Dakota State Department of Health and the Public Health Service. At that time samples of river water, mud or sand, and aquatic life were collected at four locations and analyzed for gross alpha and beta radioactivity. The sampling stations are shown in Figure 1. Biological samples included plankton, algae, insects, and minnows. The water samples contained 10-40 micromicrocuries per liter (μμc/l.) of dissolved alpha activity and 10-120 μμc/l. of dissolved beta activity. Suspended radioactivity was practically nil. River mud samples contained an average of 10 and 15 μμc/gm. of dry solids of alpha and beta activity respectively, with no observable variation between sampling stations. The biological samples showed correspondingly low concentrations of gross radioactivity. Ground water samples had dissolved alpha and beta activity in the same range as was indicated by the Cheyenne River water samples.

FIGURE 1

In 1957, a sample of slimes tails before entering the slimes tailings pond yielded gross alpha and beta activities of 590,000 and 780,000 μμc/l., respectively. Virtually all of this activity was in suspended solids, and the dissolved activities were 980 and 930 μμc/l., respectively. A liquid sample from the slimes pond indicated similar dissolved activities, with only slight suspended radioactivity. Subsequent water samples (June 1957) indicated no significant radioactivity above background in the Cheyenne River or in Cottonwood Creek, although the small direct drainage from the sands pond to this creek contained 1,400 and 1,800 μμc/l. of dissolved gross alpha and beta radioactivity. No radium analyses were made prior to the mill survey of July 1957.

In terms of radioactive waste disposal, radium is the most hazardous radioelement involved in the extraction of uranium from its ores. Only the uranium is wanted, and all of its radioactive daughters, including radium, are disposed of as waste products. Of all of these decay products, radium has by far the lowest Maximum Permissible Concentration in water.(2) Hence, the amount involved, as well as the course of its passage through the extraction process, are

of considerable interest. Of special concern is the question of how much radium becomes dissolved in the processing of ore, and where this dissolved portion goes. One of the prime aims of this study, therefore, was to perform a radium balance through the mill, and in so doing to answer the foregoing questions.

THE MILL PROCESS(3)

In brief, ore received at the mill is crushed and ground, and leached with sulfuric acid to dissolve the uranium. The coarse sands are separated and discharged to waste, and the remaining slurry, or IX feed, containing the slimes or fine solids, proceeds to ion exchange resin banks. Here the uranium is extracted from the feed solution by resin beads, is in turn stripped from the beads, precipitated, filtered and dried to form yellowcake, which is the final product of the process. This uranium concentrate is then shipped to other facilities for further refining and processing. The IX feed solution, having been stripped of its uranium, becomes the slimes tails, and is neutralized with lime and sent to the slimes pond. Figure 2 is a schematic flow diagram of the entire process.

Sampling

Loaded ore trucks dump the uranium ore into either of two ore bins from the ramp shown at the far left in Figure 2. The ore is fed to the sampling plant from these bins by a conveyor belt. A magnetic iron separator removes metal scraps, nuts, bolts, etc. In the sampling plant the ore passes over a 1-1/2 inch vibrating grizzly, then goes to a 1-1/2 inch jaw crusher, and falls on a conveyor belt. Ten per cent of the ore is removed from the belt by calibrated rotating buckets and 90 per cent goes goes directly to the mill. The 10 per cent passes over a 3/4 inch screen, goes to a 3/4 inch jaw crusher, and drops to a conveyor belt. Ten per cent (one per cent of the original ore) is removed by calibrated rotating buckets, and the remainder leaves for the mill. The retained material goes to a 1/4 inch screen and a 1/4 inch jaw crusher, and falls to a conveyor belt from which a Vezin sampler removes 10 per cent. The remainder leaves for the mill. The sample amounts to 0.1 per cent of the original ore feed, or two pounds of ore per ton of ore fed. This representative sample is assayed for its U_3O_8 content. After leaving the sampling plant the crushed ore goes to either of two 250-ton ore bins or to a 50-ton truck bin from which specification material can be taken for blending.

Grinding

The mill bins each have two bottom discharge hoppers, and one means of blending control consists of regulation of the feed rates from these four chutes. Another method of blending involves scheduling of various shippers' lots through the mill. Factors considered for primary blending purposes include the grade of ore, particle size of sand grains, slimes content, and oxidizing or reducing characteristics of the ore.

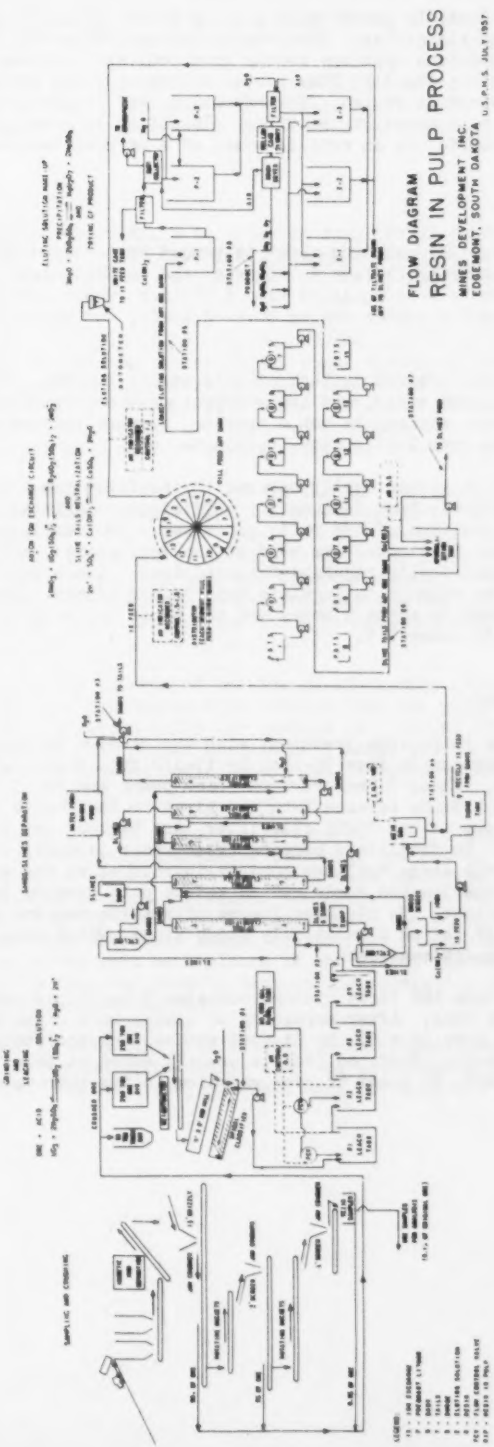
FLOW DIAGRAM
RESIN IN PULP PROCESS
MINES DEVELOPMENT INC.
EDGEMONT, SOUTH DAKOTA
U.S.P.M.S. JULY 1957

FIGURE 2

The grinding facility consists of a 4- by 8-foot rod mill in open circuit with a 42-inch spiral classifier. The feed to this grinding section is quite fine, because the sandstone ores are poorly cemented, and crushing, together with repeated handling before the mill bins causes a large portion of the ore to be broken down into individual grains. The resulting feed is easily ground to minus-12-mesh. Water is added at the spiral classifier to make a slurry. Water for the mill is obtained from an artesian well at a natural temperature of 53°C.

Leaching

The slurry from the grinding plant is pumped to a series of four wood-stave tanks, each measuring 14 feet in diameter and 14 feet high, where leaching is carried out. Each tank is equipped with a 48-inch rubber covered propellor mounted on a 4-1/2 inch diameter rubber covered shaft, and these tanks are operated in series.

Most of the ore received at this mill is easily leached. Most of the uranium is in an oxidized state, and hydrocarbons or other reducing elements are not present. Excessive sliming is not a problem, because the ore is loosely cemented and contains very little clay, bentonite, etc.

Sulfuric acid is automatically metered to the first two leach tanks from a 30,000-gallon acid storage tank in quantity sufficient to maintain pH from 0.9 to 1.4, depending upon the type of ore being processed. Two continuously recording pH meters with calomel electrodes are used at the leach tanks, and are coupled to controllers that automatically regulate the acid feed. Leaching is carried out at about 40°C, with no attempt to regulate heat in the system. The retention time in each leach tank is about 3 hours and the pH of the pulp overflow from the fourth tank is usually under 1.5.

Sand-Slime Separation

The slurry, or pulp, from leaching, with the uranium in solution, undergoes sand-slime separation in five 30-inch by 17-1/2 foot spiral classifiers and two 10-inch cyclones. Three 2-inch vertical sand pumps and two 3-inch pumps are used in this section. Sands advance from the first to the fourth classifier, and from the second cyclone to the fifth classifier, are washed, and pumped to the sands tailings pond. To facilitate pumping fresh water is added at the pump. Water recycled from the sands tailings pond is introduced at the second and fifth classifiers to wash the leached sands and to reduce pulp density for a sharper separation in the cyclone. To minimize losses of uranium-bearing solution, the spiral classifiers are set at a relatively steep slope, which results in a small pool surface and long drainage deck.

The overflow from the first cyclone contains 5 to 10 per cent solids that are minus-300-mesh in size. After screening to remove wood chips and other trash, this overflow goes to a 21- by 21-foot wooden tank for storage ahead of the ion exchange circuit. Powdered iron is usually added at this tank to adjust the solution EMF to 400, in order to keep vanadium in its tetravalent state and

thus prevent poisoning of the resin during ion exchange. The ion exchange feed has a pulp density of about 1.05, entrained solids are minus-300-mesh, pH is 1.7 to 1.9, EMF is 400, and the concentration of U_3O_8 in solution is about 1.0 gram per liter (1,000 parts per million). This pulp, with a small amount of recycle IX feed from the banks, is pumped to a small elevated constant head tank, from which it is metered by means of a weir to the distributor for the resin tanks.

Ion Exchange

Basically, the uranium is absorbed from the IX feed solution on anion exchange beads. The beads are then stripped of their uranium by an acidified nitrate solution, and the uranium is subsequently precipitated from the nitrate solution.

The resin-in-pulp (RIP) section consists of 14 rubber lined steel tanks, or banks, each containing two stainless steel baskets which are 4-1/2 by 4-1/2 feet in cross section and 5 feet high and have 30-mesh openings. The baskets hold a 10-inch bed of 20-mesh resin beads. The bank is filled with pulp slurry, and the motor-driven baskets oscillate up and down in the tank at 6.3 cycles per minute, to ensure good contact of bead surfaces with the pulp.

The distribution of solutions in the RIP section is accomplished by an ingenious device that eliminates all need for expensive valving in this section. The central unit is a distributor wheel through which IX feed solution and eluant enter the banks. The wheel is divided into 14 compartments, each connected by pipe to one bank. A pH indicator and recorder continuously gives pH of the IX feed and eluting solution, and meters indicate and control the flows of these solutions. Full flexibility of operation is obtained with the system, a detailed description of which is given by Dayton.(3)

The banks are alternated between an adsorption (loading) cycle, during which bead surfaces are loaded with uranium from the pulp, and an elution (stripping) cycle, during which the uranium is stripped from the beads by a stripping solution. In normal operation, 7 banks are on adsorption, 5 are on the elution cycle, one between these two cycles is being washed, and one bank is on standby.

The countercurrent principle is used during loading and stripping. The uranium-rich IX feed solution enters that bank containing the most completely loaded resin; fresh eluant is added to the bank where the beads have the least amount of adsorbed uranium. Pregnant eluant is taken from the bank where the beads are most heavily loaded with uranium, and the stripped IX feed leaves from the bank containing the least loaded resin. The bank which is taken off the loading cycle is the next bank placed in the circuit at the end of the stripping cycle. Solution from the seventh bank on the adsorption cycle is sent to the slimes tails neutralization tank.

Control is obtained by quick fluorimetric uranium analysis of samples taken at the first and second banks from the discharge ends of the adsorption and elution cycles. For instance, the lead bank is taken off adsorption and another

bank is added at the end of the adsorption cycle when the uranium concentration in the IX effluent to tails builds up to a predetermined level. Otherwise uranium would be lost to the tailings.

The banks are drained by means of bottom discharge ports fitted with a hollow vertical plunger. The solution level in the banks is controlled as desired via large funnels connected by means of flexible rubber hose to the hollow plunger. Excess solution is thus drained to the bank pumps. Control of the solution level provides smooth continuous flow from bank to bank, and minimizes surging and overloading any of the banks.

The beads must be hosed intermittently to keep them wet and promote drainage, once a bank has been drained of solution. The beads tend to swell and stick together when dry, but separate readily if kept wet. The bank taken off the adsorption cycle is washed to remove slimes adhering to the beads, in order not to contaminate the eluting solution. After elution, the beads are washed to remove excess nitrate. Wash water is kept to a minimum, and is recycled to the IX feed storage tank.

Eluting solution is made up in three wooden tanks. At any time, one tank is delivering fresh eluant to the RIP banks, one is receiving filtrate from yellowcake filtration, and the third, full of filtrate, is being adjusted with acid and nitrate. Fresh eluant entering the RIP section has about 56 grams per liter of nitrate ion and is acidified with sulfuric acid to a pH of 1.2.

Considerable purification and concentration of uranium results in the RIP circuit. The anion resin beads extract in the neighborhood of 99.7 per cent of the uranium in solution, but only a small fraction of the dissolved iron, vanadium and aluminum. After elution, the pregnant eluate sent to precipitation assays 10 to 20 times the uranium assay of the IX feed solution, containing 10 to 12 grams per liter of uranium, expressed as U_3O_8. This uranium-rich solution is pumped to a pregnant eluate holding tank.

Uranium Extraction

The pregnant eluate is next filtered and clarified before uranium precipitation, in order to obtain a clean concentrate uncontaminated by slimes solids carried into the pregnant eluate. Sufficient milk of lime is added to bring the pH up to 3.5 with subsequent precipitation of calcium sulfate, or whitecake, which is returned to the IX feed tank. The pregnant eluate is then clarified by a 38-frame plate and frame filter press. This preliminary filtration also controls sulfate buildup. Sulfates are also controlled by bleeding off about 10 per cent of the yellowcake filtrate to slimes tails.

The clarified pregnant eluate is next cycled through a yellowcake dust collector which strips it of dust from the yellowcake dryers. It then proceeds to one of two 12- by 14-foot precipitation tanks, where magnesium oxide is added in dry form to produce a diuranate precipitate. Enough MgO is added to produce a solution pH of 6.8. The magnesium oxide is rather slow-acting, but produces a large floc as compared to other precipitation agents. The precipitation period required from 4 to 10 hours.

All of the chemical reactions involved are indicated on Figure 2, as well as in the process description by Dayton.(3)

The precipitated slurry is pumped to one of two frame filter presses for yellowcake filtration. A layer of filter paper and a layer of nylon filter fabric are used on the press frames. After filtration the cake is washed with water and given an air blow. Below the filter presses are removable drip pans which are installed over a paddle re-pulper that extends the full length of the filter. The drip pan is removed and the filter opened; the precipitate is scraped off the frame and drops to the re-pulper, which keeps the cake fluid while it is delivered to an agitator. The concentrate is then pumped to a double-drum dryer. Here the yellowcake is splashed onto rotating drums which are interior-heated with steam. The thin layer of concentrate that forms on the drum is scraped off and drops to a hopper, from which it is drummed for shipment.

The drum dryer operates under a slight vacuum, and the exhaust is pulled through a dust separator fitted with ceramic baffles. Clarified eluate is introduced at the top of the separator and percolates down, stripping the uranium concentrate from the dryer exhaust. The scrubbed air is vented to atmosphere.

The filtrate from the yellowcake presses goes to the eluting solution tanks, where it is adjusted with acid and nitrate to make fresh eluant. Ten per cent of the filtrate is delivered to the slimes tails so as to prevent sulfate buildup.

The slimes tails from the RIP banks are sent to a neutralization tank, where lime is added to bring the pH to about 9.5. After neutralization, these wastes are delivered to the slimes tailings pond, a large lagoon located near the Cheyenne River.

THE MILL SURVEY

For the purposes of analyzing the mill process and characterizing the resulting liquid wastes, eight sampling stations were set up within the mill, and the field survey was carried out during the period July 25 through July 30, 1957. Six other sample types were collected outside the mill. Table I is a description of the various sampling stations within the mill. The other stations included:

(a) Liquid from the slimes tailings pond.

(b) Wet solids from the slimes tailings pond.

(c) Liquid from the sands tailings pond.

(d) Dry sand from the sands tailings pile.

(e) Silt and wet solids from the slimes pond outlet to the Cheyenne River.

(f) Direct liquid drainage to Cottonwood Creek.

Table I

Sampling Stations Within the Mill

Station Number	Description
1	Raw ore slurry before entering leach tanks.
2	Leached ore slurry before sand-slime separation.
3*	Sand slurry proceeding to sands tailings pond.
4	IX feed solution.
5	Loaded eluting solution from RIP circuit.
6	Slimes tails from IX banks, after adsorption of uranium, and before neutralization.
7	Neutralized slimes tails proceeding to slimes pond.
8	Yellowcake.

* Additional dilution water is added to this process stream beyond the sampling point, to facilitate pumping.

As may be seen from Table I, Sampling Stations Numbers 1, 2, 4, 5, and 8 are main mill process streams, whereas Numbers 3, 6, and 7 represent liquid waste streams.

Sampling within and outside the uranium mill was performed by personnel of the Public Health Service and the South Dakota State Department of Health. Sampling within the mill commenced on July 25, and was completed on July 30, 1957. During four days of this period, samples at each station within the mill were collected hourly for eight hours and composited. During the middle of the survey the mill samples were collected hourly for an uninterrupted 24-hour cycle and were composited into 3-hour samples. This program resulted in four 8-hour composites and eight 3-hour composite samples at each of the eight sampling stations within the mill. A full set of samples was shipped to the Robert A. Taft Sanitary Engineering Center of the Public Health Service, in Cincinnati, Ohio, for gross radioassay and for chemical analysis. A full set of identical samples was sent to

the Occupational Health Field Station, Public Health Service, Salt Lake City, Utah, for radium analysis.

One of the primary purposes of the study was to make a complete balance of all radium entering and leaving the mill in both dissolved and undissolved form, and separate radium analyses were performed on the solid and liquid phases of the samples. For these purposes, the hourly samples collected during the 8-hour periods on July 25 and 26 were composited into a single sample for radium analysis, for each mill station. The hourly samples collected over the 8-hour periods on July 27 and 30 were similarly composited. The hourly samples taken for the 24-hour period during July 28 and 29 were composited into a single sample for each station for radium analysis.

Samples of the direct drainage to Cottonwood Creek were collected on each of the latter three days of the survey. A single representative sample of dry sands was obtained from the sands pile, and a single representative sample of wet solids was taken from the slimes pond. Liquid samples were collected from the sands and slimes ponds on each of three days. One representative mud sample was collected at the slimes pond outfall to the Cheyenne River. Duplicate sets of these samples were shipped to the Public Health Service laboratories in Ohio and Utah for the analyses noted above.

A record of the tonnage of ore processed was obtained frequently during the survey from the weightometer preceding the rod mill. Weir discharge records for flow were kept regularly at the weir box preceding the RIP section, and the quantity of eluant flow was obtained from plant equipment. Water flows at the various points of addition were obtained from plant personnel, as were estimates of chemicals used in the process. These included ammonium nitrate, magnesium oxide, lime, iron, and sulfuric acid. Water from Cottonwood Creek was pumped to the sands pond during the survey period in order to provide sufficient water for use in the sand-slime separation and for diluting the sand slurry. Estimates of solids concentrations, pulp density, etc., at various points in the mill were obtained from the operating personnel.

LABORATORY PROCEDURES

Radioactivity

Radium was determined generally by coprecipitation with barium sulfate. Following pretreatment of the various types of samples to put the radium in solution, the procedure consisted essentially of evaporation with sulfuric acid, removal of polonium, coprecipitation of radium with barium sulfate, purification and alpha counting of the precipitate. Rather extensive pretreatment of the undissolved solids of the mill samples was necessary to entirely dissolve them. Those samples that contained large quantities of undissolved solids were centrifuged for separate radium analysis of liquid and solid phases. The solids were then washed with water to wash out the liquid not removed by centrifuging, and this wash water was added to the liquid portion to be analyzed for radium.

Most of the samples had large concentrations of suspended and settleable solids and the gross radioactivity determinations and suspended and dissolved

solids concentrations were performed independently on separate representative portions. This procedure eliminated the need for a large number of absolute quantitative transfers for the radioactivity determinations. A known volume of sample was filtered through a membrane filter and washed. The filter and filtered solids were removed and ashed at 600°C to constant weight. The filtrate was evaporated to dryness and ashed at 600°C to constant weight. Solids concentrations were based upon the total sample volume.

The general procedure for gross radioactivity determination is described elsewhere.(4) Dissolved and undissolved solids were analyzed separately, and a self-absorption analysis was performed for a representative sample from each station. This procedure is described elsewhere.(1)

Chemical

Chemical analysis of the various samples was performed by standard methods outlined in detail elsewhere.(5) Nitrates were determined by the phenoldisulfonic acid method, and sulfates, calcium, and magnesium by the gravimetric methods outlined. Iron was determined by the phenanthroline method, manganese by the periodate method, and chlorides by the mercuric nitrate method. The results for manganese may have some error due to interferences caused by iron and chlorides, and in future work it is planned to determine manganese colorimetrically by the ammonium persulfate method to correct for these problems.

ANALYTICAL RESULTS

During the survey period the average rate of ore processing was 517 tons per day. Table II shows the average slurry flow at each mill sampling station, as well as the specific gravity and suspended solids content. Slurry flows were based upon the observations within the plant and on computations that accounted for specific gravity of the dry solids, dry solids flows estimated from ore processed, specific gravity of the liquid phase, and observed specific gravity of the slurry. The slurry flow at Station 3 is not the entire flow going to the sands pond, as considerably more dilution water was added just following the sample collection point at this location.

Table III presents radium concentrations, dissolved and undissolved in the various mill process streams, as well as at the several locations outside the mill. These are the average results for the survey period, and are given in micromicrograms per liter of slurry (μμg/l.), and in micromicrograms per gram of dry suspended solids (μμg/g.). As can be seen, there is good agreement between the dissolved radium concentrations at Stations 2, 4, 5, and 6, and between Station 7 and the slimes pond liquid. Similarly, the concentrations of radium per gram of dry solids agree well at Stations 1 and 2, at Stations 4, 5, and 6, at Station 7 and slimes pond, and at Station 3 and sands pond (dry sands).

Gross alpha and beta radioactivity, dissolved and undissolved, for each of the mill stations are given in Table IV, in micromicrocuries per liter (μμc/l.).

Table II

Process Stream Characteristics

Station	Slurry flow Gal/min.	Specific Gravity of Slurry	Per cent Dry Suspended Solids by Weight
1	91	1.52	60
2	91	1.53	59
3	91	1.45	64
4	146	1.05	7.2
5	12	1.13	~0.07
6	142	1.05	7.2
7	151	1.05	8.2

The dissolved gross alpha and beta radioactivity show a large gain as the result of acid leaching, and, as expected, slimes neutralization results in a major reduction of the dissolved gross activities in the slimes liquid at Station 6. The results for Station 1 from Tables III and IV indicate that the radium constitutes about 16 per cent of the gross alpha activity of the ore.

Table V indicates the results of the chemical analysis of samples of the waste flows and pond contents. These results are given in milligrams per liter (mg/l.), of the liquid portion of the samples, and represent only dissolved chemicals. Nitrates are expressed as nitrate nitrogen, rather than nitrate ion. It should be noted that the sands pond was also receiving water pumped from Cottonwood Creek during the survey.

The general agreement between Station 7 and the slimes pond and Station 3 and the sands pond is evident. Nitrate nitrogen was also determined for the liquid portions of samples from other mill stations, and was essentially zero at Stations 1, 2, and 3, but was 15, 8,200, and 230 mg/l., respectively, at Stations 4, 5, and 6. Using the flows of Table II (liquid portion), the ore processing rate of 517 tons per day, and estimating chemically the nitrate ion required for exchange with uranium, it has been calculated that the nitrate ion use during the survey was about 15 pounds per ton of ore processed. This is in good agreement with the mill records.

As has been indicated, one of the primary purposes of this survey was to make a radium balance throughout the process, if possible. Table VI, based upon

Table III

Radium Concentrations

Station	μμg Ra per liter of Slurry		μμg Ra per gram of Dry Suspended Solids
	Dissolved	Undissolved	
1	98	605,000	650
2	2,150	640,000	710
3	114	163,000	170
4	2,450	273,000	3,640
5	2,290	2,050	~2,800
6	2,450	288,000	3,760
7	350	250,000	2,930
8	--	--	150
Liquid from Slimes Pond	270	--	--
Liquid from Sands Pond	8	--	--
Drainage to Cottonwood Creek	17	--	--
Dry Sands	--	--	150
Solids from Slimes Pond	--	--	2,470
Slimes Pond Outlet Ditch	--	--	58

Table IV

Gross Radioactivity

Station	Gross Alpha, μμc/l.		Gross Beta, μμc/l.	
	Undissolved	Dissolved	Undissolved	Dissolved
1	3.66×10^6	1.71×10^3	4.79×10^6	1.02×10^3
2	2.86×10^6	473×10^3	3.96×10^6	613×10^3
3	1.08×10^6	3.08×10^3	0.992×10^6	2.05×10^3
4	0.880×10^6	388×10^3	0.962×10^6	555×10^3
5	0.010×10^6	$2,930 \times 10^3$	0.017×10^6	$5,740 \times 10^3$
6	0.810×10^6	94.5×10^3	0.866×10^6	47.5×10^3
7	0.725×10^6	1.08×10^3	0.610×10^6	1.28×10^3

Table V

Quality of Waste Flows and Ponds Contents

Sampling Station	mg/l. of Material Dissolved in Liquid Sample						
	Sulfate	Chloride	Calcium	Magnesium	Iron	Manganese	Nitrate Nitrogen
Slimes to Tails (Station 7)	2,330	205	730	75	0.14	~0	500
Liquid from Slimes Pond	2,190	200	820	80	~0	1.3	460
Sands to Tails (Station 3)	2,180	240	570	120	Trace	7.0	~0
Liquid from Sands Pond	1,970	275	440	150	~0	~0	~0
Drainage to Cottonwood Creek	1,090	170	360	65	~0	~0	Trace

Table VI

Radium Balance

Station	Milligrams per Day of Radium		
	Undissolved	Dissolved	Total
1	301	0.049	301
2	316	1.06	317
3	81	0.057	81
4	217	1.95	219
5	0.1	0.150	0.25
6	223	1.90	225
7	225	0.316	225

the data of Tables II and III, presents the radium balance for the mill, and indicates the various paths by which specific quantities leave the mill. The acid leaching process dissolves a certain amount of radium as well as the uranium. Some of the suspended radium remains tied up with the sands (about 170 μμg Ra per gram of dry sands) and is discharged to tails with them at Station 3. A portion of the dissolved radium (150 microgram per day) present at Station 4 goes with the loaded stripping solution at Station 5, although the data of Table III indicate that dissolved radium is not extracted or concentrated by the ion exchange section. As will be seen, virtually all of this carried over radium becomes a part of the yellowcake that is shipped out of the mill. The neutralization of the slimes tails with lime results in precipitation of a substantial fraction (about 85 per cent) of the radium dissolved in the liquid at Station 6, as shown by both Tables III and VI.

As a check on the data of Table VI, the radium content of the ore can be estimated within reasonable limits of error. On the basis of 517 tons per day of ore that assays 0.20 per cent U_3O_8, and assuming radioactive equilibrium of radium with the uranium, it has been calculated that 270 milligrams per day of radium enter the mill with the ore. This is in satisfactory agreement with the totals of Table VI.

Further computations indicate that virtually all of the radium present at Station 5 (150 micrograms per day) becomes a part of the final uranium concentrate. Assuming 96 per cent over-all recovery of uranium, and noting that the yellowcake is approximately 75 per cent uranium as U_3O_8[3], a calculation indicates that roughly 1.2 tons of yellowcake are produced per day. Using the radium

concentration of 150 μμg per gram of dry solids from Table III it results that about 160 micrograms per day of radium leave the mill with the yellowcake. This is quite close to the figure of 150 micrograms per day for Station 5 (Table VI).

From Table VI it is evident that about 300 milligrams per day of radium enter the mill. About 80 milligrams per day leave with the sands at Station 3, and 220 continue through the process to the slimes pond. Some 60 micrograms per day of dissolved radium are discharged to the sands pond and about 300 micrograms per day of dissolved radium go to the slimes pond. Using the data of Tables III and IV, at Station 7, it is seen that of the dissolved gross alpha activity of 1,080 μμc/l., about 30 to 35 per cent is due to dissolved radium.

In similar fashion rough balances of the gross alpha and beta radioactivity can and have been made. They need not be reproduced here, as radium is the specific radioelement of major interest, and its balance has been shown. It may be of interest, however, to indicate that the undissolved radium varies between 15 and 38 per cent of the undissolved gross alpha activity at Stations 1 through 7, while dissolved radium constitutes from 0.1 to 35 per cent of the dissolved gross alpha activity. Details of these figures can be obtained by a simple computation using the data of Tables III and IV.

It is also of interest to note that, although dissolved radium is not concentrated in the ion exchange section, dissolved gross radioactivity is. Table IV indicates that the dissolved gross alpha and beta radioactivity were concentrated by a factor of 8 to 10 in the RIP section, or between Stations 4 and 5. This is in good agreement with the estimated concentration of uranium by a factor of 10, as quoted by mill personnel and by Dayton.(3)

SUMMARY AND CONCLUSIONS

For the ore that was being processed at the uranium refinery under study, about 17 per cent of the gross alpha activity of the ore was due to the presence of radium. At the ore processing rate of 517 tons per day, the liquid and solid wastes from ore processing contained about 0.6 milligrams of radium per ton of ore processed, or a total of approximately 300 milligrams per day. By far the bulk of this radium, or some 99.8 per cent, remained in undissolved form throughout the process, and was effectively retained in the sands and slimes tailings ponds.

About 0.2 per cent of the radium that entered the mill with the ore either left the refinery in the final uranium concentrate or was delivered in dissolved form to the slimes or sands pond. Specifically, it has been estimated that some 310 micrograms per day (μg/day) of dissolved radium went to the slimes pond, about 60 μg/day went to the sands pond, and about 150 μg/day of radium left the mill in the dried yellowcake.

The radium content of the dried sands was about 150 micromicrograms per gram of sands, while the radium content of dried slimes solids was 2,500 or more micromicrograms per gram. Sands accumulation was in the neighborhood of 440 tons per day, while slimes accumulated at a rate of about 80 tons per day. Effective

retention and confinement of these solids and of the tailings pond liquids resulted in practically no radium leaving the plant site during the survey, except for that contained in the uranium concentrate that was drummed and shipped. The very small amount of direct drainage to Cottonwood Creek was from the sands pond and contained little radium or nitrate nitrogen.

Consideration of the data of Table III indicates that some of the radium becomes dissolved during the acid leach process (see data for Stations 1 and 2). However, radium was not concentrated by the RIP section - the concentration of dissolved radium in the loaded stripping solution (Station 5) was essentially no different from that in the IX feed (Station 4) or the slimes to tails (Station 6). The opposite occurred as regards gross alpha and beta radioactivity: the gross alpha activity in the IX feed was 3.88×10^5 μμc/l.; in the loaded stripping solution it was 29.3×10^5 μμc/l., or greater by a factor of 7.5, and in the slimes to tails it was reduced to 0.95×10^5 μμc/l., or by a factor of 4.1 as against the IX feed. The gross beta activity behaved similarly - the loaded stripping solution had a concentration 10.3 times that of the IX feed, while the slimes to tails showed only 8.6 per cent of the dissolved beta activity of the IX feed. Hence, the data indicate that although uranium was concentrated as usual by the RIP section, radium was not concentrated here.

Slimes neutralization before discharge to the tailings pond reduced the dissolved radium concentration from 2,450 to 350 μμg/l., or by about 85 per cent. Liquid from the slimes pond showed a comparable concentration of about 270 μμg/l. of dissolved radium. Based on the data for Station 7, about one-third of the dissolved gross alpha activity in the slimes pond liquid was due to radium, at the mill studied.

The nitrate nitrogen concentration in slimes pond liquid was 460 mg/l. While this is probably of no public health significance in this case, such concentrations could be of considerable importance at other locations where public water supplies are involved. These quantities of nitrogen could also provide sufficient nutrient material to result in undesirable blooms of algae and other biota in streams and reservoirs. Shortly after the field survey was completed the use of sodium nitrate in place of ammonium nitrate was instituted at the refinery, but it is doubtful that this would appreciably alter results presented here.

Waste Disposal Practice

As has been noted, there has been quite careful control of wastes at the Mines Development Company refinery at Edgemont, South Dakota. Sand and slime solids have been effectively confined at the plant site, and liquid waste releases have been at a minimum.

Waste disposal from this mill and from mills employing a similar process should continue to be carefully supervised, and should in any instance be based upon available knowledge of downstream water uses and of the fate of the wastes in the water environment. The sand and slimes solids should generally be effectively retained and confined, as they contain considerable radium and other

radioelements. Their release to a stream would result in long term contamination of the watercourse, and they could significantly contaminate equipment at downstream domestic or industrial water treatment plants.

Liquid wastes from the slimes ponds at this and similar mills should be released to surface waters only in accordance with existing regulations,[2][6] and on the basis of detailed information as to local downstream water uses. This liquid, free of suspended solids, contains considerably more than allowable concentrations of radium and of nitrate nitrogen. At the uranium refinery studied, such control is especially important in view of the zero flows and extended low flow periods that occur in the Cheyenne River. As has been noted, these flows are such that little or no dilution of the effluent would occur for extended periods of time. Any susceptible ground water supplies located near the slimes tailings ponds of such refineries should be tested periodically for nitrate nitrogen and radioactivity content, as infiltration of the slimes pond liquid into the ground water may occur at the pond.[7]

As regards measurement of radioactivity in the effluent from the slimes tailings pond at Edgemont, South Dakota, it appears reasonable to suppose from these studies that the dissolved radium content will generally be in the neighborhood of 30 per cent of the dissolved gross alpha activity. It seems feasible, for routine measurement purposes, therefore, to analyze these samples for dissolved gross alpha activity and apply the factor of 0.30 to compute dissolved radium. An occasional analysis for radium itself will then serve as a check, and refine the percentage figure to be used. In this way, the costs and labor of sample analysis for routine control can be minimized, while providing adequate control. It may be that this percentage figure would also prove adequate within reasonable limits of error for application at other mills using the same process, but this cannot be known until further studies are carried out.

Increases or decreases in the rate of ore processing at the Edgemont, South Dakota, refinery, or changes in the U_3O_8 content of the ore, should be proportionately reflected in many of the waste quantities. This applies particularly in the case of radium. Thus, assuming no important process change, a capacity increase to, say, 750 tons of ore per day, at 0.20 per cent U_3O_8 should result in about 440 milligrams per day of radium entering the mill, and about 460 micrograms per day of dissolved radium entering the slimes tailings pond. Any basic change in the type of ore received would, of course, affect these figures, especially the latter.

It is not intended here to generalize very much beyond the local situation studied. Each uranium refinery waste disposal problem is individual, and must be interpreted in terms of specific local water uses such as domestic water supply, irrigation of croplands, recreation, etc., as well as in terms of the specific waste characteristics and mill process. Available dependable dilution in the receiving waters is also a critical factor in control of these wastes. No two such problems are likely to be precisely the same, and in the dual interest of radiological safety and economy, such waste disposal problems should each be carefully analyzed as individual cases insofar as necessary. It is hoped that the information presented herein will provide some insight into the waste disposal problems at acid leach resin-in-pulp mills generally, but the data should not be applied to other mills without considerable caution.

FUTURE RESEARCH

In cooperation with the U. S. Atomic Energy Commission and the uranium milling industry, it is planned to make similar studies on a reduced scale at other resin-in-pulp mills. These studies will involve fewer samples and will be designed to provide added information as well as to support or modify the data presented here. Other types of refining process will be studied in similar fashion, so as to develop as general a guide as possible to the problems of waste disposal from the industry.

In addition, studies of the fate of these wastes in the water environment will be continued by the Public Health Service, and will involve, for example, investigations of uptake of radium and other elements by the aquatic biota, effects of radioactive suspended solids in water treatment plants, and other specific questions.

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NATURE OF WASTES IN FUEL ELEMENT MANUFACTURE

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The Babcock & Wilcox Company has been producing fuel elements in its Nuclear Facilities Plant at Lynchburg, Virginia, since June 1956. These elements have been produced for research reactors, the Materials Testing Reactor, and the Engineering Test Reactor, at Arco, Idaho, and for large scale power plants such as the nuclear submarines. We will be fabricating fuel elements for the Nuclear Merchant Ship and the Consolidated Edison plant being built at Indian Point, New York. The process of manufacturing fuel elements for research and power reactor service involves the use of uranium, either as a pure metal, as an oxide or as an alloy. Manufacturing operations involved in this type of fabrication may include melting, casting, rolling, machining, and metal cleaning by the use of acids. Any of these operations may produce varying quantities of uranium-bearing wastes in the form of alloy scrap, liquids, contaminated pieces of metal, paper or rags, and in the event that oxides are used, a certain amount of airborne dust is generated. Precautions must be taken to prevent hazards to health and safety and to comply with applicable regulations. Moreover, in all cases where the waste material contains significant quantities of highly enriched uranium, the strategic importance of the metal as well as its high monetary value dictate the necessity of reprocessing the waste in order to recover the uranium metal. Even in the case of lower enrichments, the quantity of uranium involved may justify recovery. In this presentation, we are not discussing uranium recovered from

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In 1930 I started to work in the laboratories of the Babcock & Wilcox Company and have been concerned in that company principally with problems having to do with the metallurgy and chemistry involved in power boiler fabrication. For the past three years I have been in charge of the company's Nuclear Facilities Plant operation at Lynchburg, Va.

waste products for reprocessing, but rather the uranium-bearing wastes that cannot be economically recovered and must be disposed of.

Radioactive Characteristics of Uranium

Uranium is a radioactive metal and emits alpha, beta and gamma radiation. The majority of the energy released is in the form of alpha emission. Fortunately, alpha radiation has an extremely low penetrating power and is, in general, stopped by as little as a sheet of paper. The levels of beta and gamma radiation from uranium as used in our plant are so low that the external effects thereof do not involve a significant hazard. Thus, the handling of uranium wastes does not represent a significant problem insofar as external exposure to radiation is concerned. However, it is necessary to provide extensive precautionary measures to prevent the accidental ingestion or inhalation of significant amounts of uranium in the form of airborne dust. If inhaled in significant quantities, the uranium may localize in various parts of the body and cause injury.

Uranium used by fuel element manufacturers has an extremely low specific activity when compared with fission products from a fission reaction. For example, it requires 16.7 kilograms of uranium having an enrichment in excess of 90% to produce an activity equivalent to 1 curie. On the other hand, strontium 90, a long-lived fission product, will exhibit the same 1 curie of activity based on a quantity of only 5 milligrams of this material. Rules and regulations determining the disposal measures for radioactive wastes are generally based on the specific radioactivity of the material involved and the unit of radioactivity is the curie which is a measure describing the number of atoms decaying per second. It is important to realize that a relatively low degree of radioactivity is involved in the handling of uranium waste products resulting from fuel element manufacture as compared with similar problems

encountered in reactor operation. This low specific activity of uranium greatly decreases the problems of waste disposal in fuel element manufacture as compared to the problems encountered with products of fission reaction.

Part 20 of the Federal Regulations, "Standards for Protection Against Radiation", permits the disposal of 26 microcuries of soluble uranium in each 1,000 gallons of plant liquid waste predicated on the limits established for general public consumption. Twenty-six (26) microcuries of enriched uranium is equivalent to 0.4 grams of material. It is, therefore, evident that only an exceedingly small amount of uranium (that is, one tenth of a part per million by weight) may be disposed of in liquid wastes.

Types of Waste

In general, the types of uranium-bearing waste which are generated in the course of fuel element manufacture and eventually have to be disposed of as radioactive waste fall into the general classifications of liquids, combustible solids, non-combustible solids. The quantities of the different types of wastes produced in our operation in 1957, together with their physical and chemical characteristics, are presented in the following description.

1. Liquid Waste

The bulk of liquid waste results from metal cleaning and etching operations and consists primarily of the final rinse solutions. The initial undiluted volume of solutions of this type generated in our 1957 operations was approximately 5,000 gallons. This solution is a dilute nitric-hydrofluoric acid solution containing small amounts of dissolved uranium along with some non-radioactive metals. In general, the level of radioactivity contained in this type of liquid waste is small enough that it can be readily diluted by other non-contaminated liquid waste and discharged to the river which borders our plant site. In order to assure that no liquid containing radioactive material

in excess of the level permitted by Federal Regulations is accidentally discharged, all of such liquids are first diverted to a series of holding tanks and monitored by the Health & Safety Section prior to release to the river. In this manner we do not only assure adequate control of the waste disposal problems but also effectively prevent the accidental loss of significant quantities of uranium metal.

2. Solid Non-combustible Waste

Since uranium and alloys containing uranium tend to pick up oxygen and other gases from the air during elevated temperature processing, it becomes necessary that all such processing be conducted under conditions that completely exclude exposure of the metal to the atmosphere. In general, this sealing is accomplished by enclosing the uranium-bearing ingot in a metal sheath during the forging and hot rolling operations. In the process of hot working the metal within its sheath, the sheath, or jacket as it is frequently called, becomes contaminated with uranium and must be handled as a radioactive waste material. In the course of our 1957 operations some 7,000 pounds of contaminated copper and steel jackets were generated. Although the amount of contamination rarely exceeds 3 parts of uranium per million parts of steel, or about 1/2 part of uranium per million parts of copper, it is considered necessary to decontaminate the metal prior to its disposal. Uranium contamination is removed from both copper and steel by the use of acids and, after proper cleaning, the copper and steel jackets are carefully monitored by Health & Safety personnel prior to release as non-radioactive scrap. The amount of uranium recovered in this manner is generally too small to warrant economic reclamation and the waste liquids are diluted, neutralized, and transferred to the holding tanks. Prior to the release of this liquid waste from the holding tanks to the river, the Health & Safety Section again monitors the level of radioactivity.

3. Miscellaneous Non-combustible Wastes

In the course of handling uranium-bearing materials in the plant, certain equipment such as asbestos gloves, refractory crucibles, stirring rods, and other materials that cannot readily be burned become contaminated by uranium. In the course of our operations during 1957 we produced in the order of 5,000 pounds of this type of waste. An analysis based on radiation surveys of this waste material indicated that the total uranium content in the 5,000 pounds involved was significantly less than 200 grams of uranium. Waste materials of this type are generally packed in 55-gallon steel drums and returned to disposal facilities.

4. Combustible Waste

In the process of handling uranium metal and, more particularly, uranium oxides, it is often desirable to cover work surfaces, tables, floors, and certain parts of equipment with paper or plastic. This is done in order to assure that the working surfaces are not contaminated with small particles of uranium that may be difficult to remove. The paper or plastic used as a covering material then becomes contaminated with uranium particles and must be handled with proper Health & Safety precautions. It is our practice to store this contaminated paper and plastic in 55-gallon drums and in the course of our 1957 operations 50 such 55-gallon containers were accumulated. In this type of application the amount of uranium involved in individual instances can vary widely; however, our analyses to date indicate that the total amount of uranium involved in this type of waste was of the order of only a few grams. In order to reduce the bulk of this waste, we are equipped to burn this material at our plant site and the ash resulting from the combustion of this material is salvaged for recovery of the contained uranium.

In the near future, The Babcock & Wilcox Company plans to fabricate

fuel elements containing thorium dioxide for the Consolidated Edison Company of New York. It is anticipated that waste materials from handling thorium will be of the same magnitude as those we have experienced with uranium.

RADIOACTIVE WASTES FROM REACTOR OPERATION

Prepared by Members of The Argonne National Laboratory Staff

I. ORIGIN OF RADIOACTIVE WASTES

The radioactive wastes stemming from nuclear reactor operation are produced by two general processes: (1) fission of the fuel, and (2) neutron activation of the coolant and the impurities contained therein, and the structural components.

In the fission process, the atomic composition of the fuel is bombarded by neutrons which, in addition to the production of heat, yields potential radioactive contaminants in the form of gases and solids.

The neutrons which do not effect collisions with the fissionable material travel relatively long distances and thus give rise to most of the radioactive wastes in reactor plants. They are absorbed by the coolant and contained impurities, and by the structural components, to form radioactive isotopes of the absorbing elements.

The degree to which both types of wastes are produced and controlled varies with reactor classification in terms of application, fuel coolant and/or moderator. Reactors are generally classified into two broad categories: heterogeneous and homogeneous. The heterogeneous reactors are typified by a core lattice of solid fuel elements clad with a structural material (aluminum, zirconium, stainless steel) to protect the fuel from the coolant (water, liquid metal, organic, air) and to contain the fission products. All heterogeneous reactor systems feature elaborate electronic devices designed to detect a clad rupture sufficiently in advance to permit removal of the failed element before significant amounts of fission products are released into the coolant stream. By-pass coolant purification systems provided for the removal of corrosion and impurity activities are also capable of retaining the fission products.

Homogeneous reactors are readily identified as containing fuel dispersed in the coolant-moderator which is cycled through an integrated primary and secondary heat removal system. The entire coolant system is the primary container of induced activities and fission products. As a consequence, the reactor complex is enclosed within a structure designed to contain the high-level activity in the event of a system failure.

The specific problems peculiar to the operation of the various types of heterogeneous and homogeneous reactors are treated in detail in the following sections.

II. HETEROGENEOUS REACTORS

A. Water-Cooled

The heterogeneous reactors in operation in this country are predominantly water-cooled. The applications range from basic research, materials testing, and isotope production, to heat generation units for central station power plants.

During operation the major source of radioactive waste is the short-lived isotopes produced by nuclear reactions in the bulk coolant (see Table I). This activity, which is a function of reactor power level, approaches a few microcuries per milliliter. If nondeaerated water is used, argon-41 activity in the order of a microcurie per milliliter may build up also.

Table I

WATER ACTIVATION PRODUCTS

Radioactive Isotope	Half-Life	Nuclear Reaction
N^{16}	7 sec	$O^{16}(n,p)N^{16}$
N^{17}	4 sec	$O^{17}(n,p)N^{17}$
O^{19}	30 sec	$O^{18}(n,\gamma)O^{19}$

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The impurity activation products are of prime concern. The principal ions of interest are shown in Table 2. Reactor systems which contain aluminum have sodium-24 as the principal activity. Cobalt-58 and cobalt-60 are formed from the nickel and cobalt in stainless steel used for construction. To these induced activities may be added fission product activities introduced into the coolant by fuel element failure. These fission products seldom markedly increase the difficulty of the waste problems.

Table 2

IMPURITY ACTIVITIES IN WATER

Radioactive Isotope	Half-Life	Nuclear Reaction
Na^{24}	15 hr	$Al^{27}(n, \alpha) Na^{24}$
Al^{28}	2 min	$Al^{27}(n, \gamma) Al^{28}$
A^{41}	110 min	$A^{40}(n, \gamma) A^{41}$
Mn^{56}	2.58 hr	$Fe^{56}(n, p) Mn^{56}$
Co^{60}	5.2 yr	$Co^{59}(n, \gamma) Co^{60}$
Co^{58}	71 d	$Ni^{58}(n, p) Co^{58}$
Fe^{55}	2.9 yr	$Fe^{54}(n, \gamma) Fe^{55}$
Fe^{59}	45 d	$Co^{59}(n, p) Fe^{59}$
Cr^{51}	27 d	$Cr^{50}(n, \gamma) Cr^{51}$

In all types of experimental water-cooled reactors, and in most of the production reactors some air cooling is used. For instance, air may be used for cooling of shielding, instrument tubes, or experimental facilities. Therefore, in addition to a liquid disposal system there must be a gaseous disposal system. The 1% of argon-40 present in the atmosphere (see Table 2) gives rise to argon-41. The coolant air containing argon-41 is usually discharged through stacks. The gaseous activity discharged through the stack of the

Experimental Boiling Water Reactor is, less than one curie per day. During 1958 CP-5 discharged approximately 40 curies/day of activity, mainly argon-41.

1. Hanford Production Reactors

The Hanford production reactors are designed for low pressure, single-pass water cooling. The large size of the reactors demands enormous quantities of water. Present pretreatment of the raw water, which is obtained from the Columbia River consists of coagulation of solids and removal by settling and filtration by rapid sand filters, chlorination, addition of corrosion inhibitors, and pH adjustment.

Effluent water from the reactor flows by gravity to retention tanks for a hold-up period of approximately an hour to permit partial decay of the short-lived radioisotopes. The ultimate disposition of the outfall is governed by the prevailing level of activity. Under normal circumstances, the discharge is through piping which extends well into the main current of the river. Upon occasion the retention basin monitoring system may register an abnormally high level of radioactivity (for example, fission products released by a clad rupture). In this event, the contents may be pumped to a storage basin where the radioactive materials are retained by percolation of the water into the ground.

The process ensuing a fuel element failure is as follows. The coolant penetrates the clad defect and promotes oxidation of the uranium. The formation of low-density oxides continues, with consequent progressive swelling and distortion of the cladding. Ultimately, the clad ruptures and the oxide particles are released into the coolant stream.

Although the detection systems employed function to prevent catastrophic failure of the fuel elements, the oxide particles that are released from a defect entrain the majority of the fission products. However, since the particles are heavier than water they are readily separated from the coolant by settling in the retention basin.

2. Materials Testing Reactor

The Materials Testing Reactor (MTR) uses a recirculation water system. The feed water is pretreated by ion exchange operations to reduce impurities to < 1 ppm. The total flow through the core and reflector is about 22,000 gallons per minute. The major materials of constructipn exposed to the water are stainless steels, cast iron, aluminum alloys, and beryllium.

During the first two years of operation of this reactor very minor fission product activities were released to the coolant system. The activity near the piping, after nitrogen-16 had decayed, was about 50 to 100 mr/hr during operation, and 10 to 20 mr/hr during shutdown. Corrosion product and other activities identified included aluminum-28, sodium-24, cobalt-60, and trace amounts of nine other isotopes. Beginning in June, 1954 a series of fuel element ruptures occurred which gradually increased the activity levels from the 50 to 100 mr/hr to as high as 3 r/hr. In October, 1954 a by-pass cation resin bed (125 cu ft) was installed and operated at a capacity of 1000 gallons per minute of the bulk coolant. As uranium-235 and fission products were removed from the system, the activity decreased slowly to their original values. This experience indicates that while fuel ruptures produce a marked increase in coolant activity, the normal levels can be restored with the aid of relatively simple cleanup procedures.

3. Pressurized Water Reactors

In order to prevent boiling on the fuel assembly heat transfer surfaces some water reactors are operated at pressures ranging up to 2000 pounds per square inch. The Naval Reactor Test Facility, Submarine Thermal Reactor, and the Pressurized Water Reactor are examples of this type. Increased temperatures reflect an increase in activated corrosion products due to the higher corrosion rates.

Data obtained at the Naval Reactor Test Facility during operation are shown in Table 3.

Table 3

TYPICAL RADIOCHEMISTRY OF COOLANT IN NAVAL REACTOR TEST FACILITY[a]

Nuclide	Half Life	Specific Activity μc/ml	Source
N^{16}	7.3 sec	100	O^{16} in H_2O
N^{17}	4.1 sec	800[b]	O^{17} in H_2O
K^{38}	7.7 min	5×10^{-2}	?
A^{41}	1.8 hr	4×10^{-2}	Air in H_2O
F^{18}	1.9 hr	4×10^{-2}	
Mn^{56}	2.6 hr	0.5×10^{-2}	Steel
Cu^{64}	12.8 hr	3×10^{-4}	17-4 pH steel
Na^{24}	15 hr	1×10^{-3}	Na in H_2O
W^{187}	24 hr	3×10^{-3}	Stellite
Fe^{59}	45 d	1.1×10^{-5}	Steel
Ta^{182}	111 d	0.6×10^{-5}	Steel
Co^{60}	5.3 yr	2.5×10^{-5}	Steel

[a] Welinsky, I. H., Cohen, P. and Seamon, J. M., "Chemistry of a Pressurized Water Nuclear Power Plant: A Review of Two Years Operating Experience," Proc. of the Amer. Power Conf. Vol. XVIII, 559-570 (1956).

[b] neutrons/(sec)(ml)

The specific activity of the wastes anticipated from the PWR is given in Table 4.

Table 4

RADIOACTIVE WASTES AT PWR[a]

Type	Monthly Amount	Specific Activity,[b] μc/ml
Reactor plant effluent	< 16,000 gal	7.6[c] 1.7[d]
High-solid-content liquids	7,500 gal	< 0.27
Spent resin	36 cu ft	10^4
Combustible wastes average maximum	3,000 lb 10,000 lb	Variable

[a] J. R. La Pointe, Nucleonics, 15 (5), May, 1957

[b] Total activity of liquids and gases, ~20 curies/day

[c] Volatile

[d] Nonvolatile

4. Boiling Water Reactors

In a direct-cycle boiling water reactor the generated steam flows directly to turbines. The condensate is returned to the reactor tank. The Experimental Boiling Water Reactor (EBWR) at Argonne and the Dresden and Vallecitos reactors are examples of this type. These reactors produce the same kinds of wastes as the pressurized water reactor with perhaps some added importance given to gaseous wastes.

The predominant gaseous activities from the EBWR are xenon-138 and krypton-88 which are produced as fission products from uranium contamination on the outside of the fuel elements. These gases are discharged to the atmosphere. The normal daily discharge rates at 20-MW operation are 0.2 to 0.4 curie of xenon-138 and 0.005 to 0.01 curie of krypton-88. During a test period when an oxide type test fuel element was deliberately defected, the discharge of these isotopes rose to 1.0 and 0.08 curie per day, respectively.

Each time the reactor vessel is opened the EBWR system is filled with water and the system is given a hydrostatic test. Following the test the excess water is bled off. This procedure has produced about 30,000 gallons/year of liquid wastes with a specific activity of about 5×10^{-5} μc/ml. This volume has been added to the regular laboratory waste disposal system at Argonne with no difficulty. Table 5 lists the principal activities which have been found in the reactor water during operation.

Table 5

PRINCIPAL ACTIVITIES FOUND IN REACTOR WATER DURING OPERATION

Isotope	Half-Life	Concentration cpm/ml	Abundance % of Total gamma Activity
Na^{24}	15 hr	3×10^5	76
Mn^{56}	2.6 hr	8×10^4	20
Co^{58}	71 d	8×10^3	2

There has also been collected during the past year at EBWR a little over 100 cu ft of solid wastes containing about 100 millicuries of activity per cu ft. Ten to twenty per cent of this volume is confined to spent ion exchange resin beds, with the balance ranging from wiping rags to assorted debris. Radiation levels of the spent resin beds have averaged about 10 r/hr at contact.

5. Heavy Water Reactors

Recirculated heavy water is used as moderator and/or coolant in such reactors as the CP-5 at Argonne, the NRX at Chalk River and the Savannah River reactors.

In addition to nitrogen-16 and nitrogen-17, and oxygen-19 formed from the oxygen, tritium is formed in the coolant by the irradiation of deuterium $\left(H^2 (n,p) H^3\right)$. The presence of the first three of these isotopes requires the use of shielding for the circulation equipment outside the reactor in normal operations but the isotopes have such short half-lives that they constitute no problem during shutdown. The amount of tritium with its long half-life (12 years) continues to increase with reactor operation for many years so that eventually precautions must be taken to avoid exposure of personnel.

The usual problems of corrosion product activation are encountered in the heavy water system and are handled in the same way as in light water systems, i.e. a side stream of the heavy water is circulated through filters and a previously deuterized ion exchange resin. The ion exchange resin accumulates most of the waste activity. The spent resin beds are removed to appropriately shielded storage facilities.

B. Liquid Metal Cooled

The liquid metals, sodium and NaK, are employed as coolants in several reactor systems. Examples of this type of reactor are the Experimental Breeder Reactor (EBR-I), the land-based prototype of the Submarine Intermediate Reactor (SIG), and the sodium-cooled graphite-moderated reactor.

The nuclear characteristics of the principal activities found in liquid metal cooled reactors are shown in Table 6. The radiation from activated impurities in commercial sodium is usually negligible compared with the bulk sodium. The radioactivity associated with sodium decays by a factor of 10 every two days.

Table 6

PRINCIPAL ACTIVITIES IN LIQUID METAL COOLANTS

Product Nuclide	Product Half-Life	Nuclear Reaction
Na^{24}	15 hr	$Na^{23}(n, \gamma) Na^{24}$
Na^{22}	2.6 y	$Na^{23}(n, 2n) Na^{22}$
Rb^{86}	19.5 d	$Rb^{85}(n, \gamma) Rb^{86}$
Sb^{124}	60 d	$I^{127}(n, \alpha) Sb^{124}$

Because of the high activity level of the sodium coolant and the fact that uranium metal is not appreciable attacked by sodium, fuel element failures are not easily detected by activity level changes. It is probable therefore that fuel element failures will go undetected for long periods of time. However, it is not expected that the fission products will be released into the sodium to any marked degree.

Disposal of waste sodium or NaK from a reactor is complicated not so much by the radioactivity content as by the vigorous chemical reaction of the metals with water and oxygen and also with carbon dioxide. There are several possible methods of disposal all of which aim at forming water soluble compounds of the metals. Water, steam, alcohols and liquid ammonia may be used for this purpose.

C. Air-Cooled Reactors

Two reasonably large air cooled research reactors are presently in operation, one at Oak Ridge and the other at Brookhaven. The Brookhaven

reactor shown in Fig. 1 features a primary air coolant system and a secondary system which maintains the temperature of the concrete plenums and exit air ducts below 212F. Low efficiency inlet air filters are used to remove dust. Following the reactor the air goes through air coolers and filters. These high efficiency air filters, AEC No. 1, have a very long life since the air leaving the reactor is essentially free of particulate matter. The original filters are still in use at Brookhaven after six years of operation. The radiation level of the exit filters is approximately 10 to 15 r/hr at contact. Some 4000 curies of argon-41 are discharged through the stack. Even in the case of fully air-cooled reactors the quantity, while troublesome insofar as instrumentation is concerned, causes no problem in regard to personnel.

D. Organic Moderated Reactors

The Organic Moderated Reactor, OMRE, uses polyphenyls as the moderator and coolant. The wastes accumulated from normal operations are primarily decomposition products resulting from irradiation of the organic material. These products contain negligible quantities of radioactive containments and are readily disposed of by burning.

III. HOMOGENEOUS REACTORS

A. Aqueous Homogeneous Reactors

Maintenance of the coolants of homogeneous reactors requires the separation of the fission products from the fuel-coolant mixture external to the reactor. The waste problems are equivalent to those experienced in the operation of a chemical processing plant.

The entire system complex becomes contaminated with deposited activities to a degree many orders of magnitude greater than heterogeneous reactors; thus large volumes of relatively low-level waste decontamination solutions are produced during clean-up operations. The radioactive fission products, krypton, xenon, and iodine are continuously removed from the coolant along with hydrogen, oxygen and steam. This gas mixture is cooled,

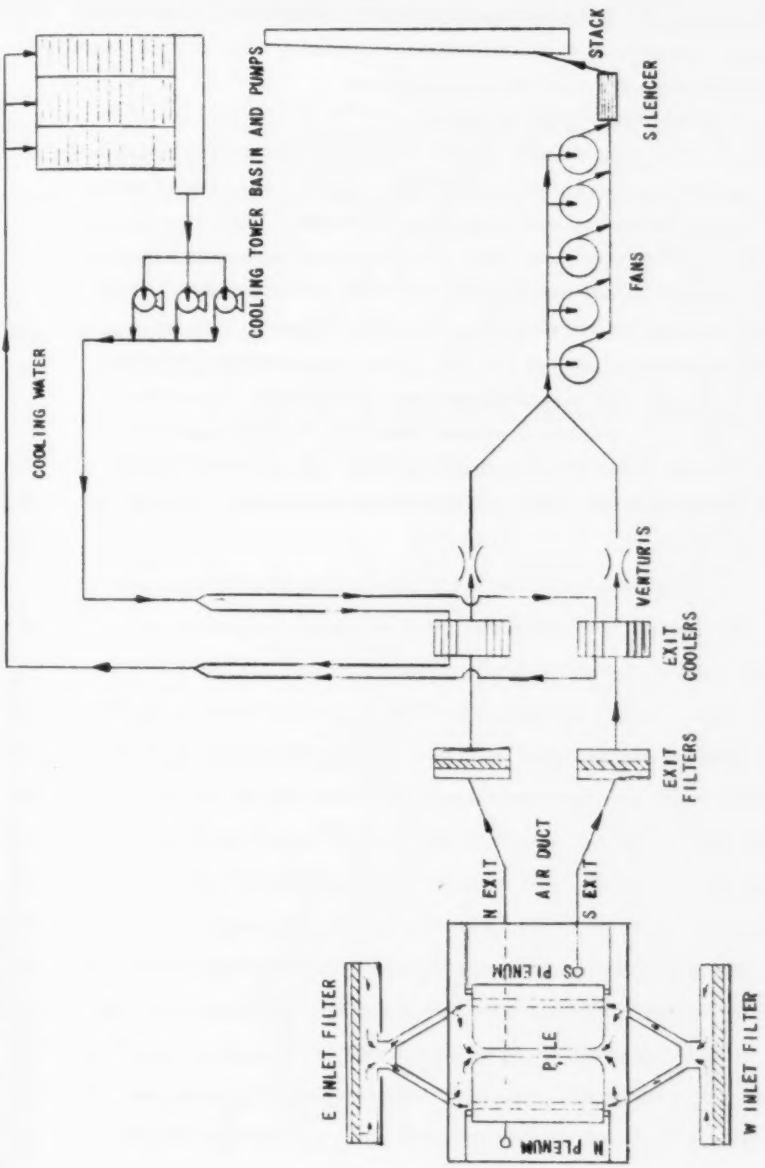

FIG. 1
PRIMARY AIR COOLANT SYSTEM OF THE BNL REACTOR

the steam condensed and the hydrogen and oxygen recombined. The fission products are physically absorbed on activated charcoal. The effluent gases are discharged through a stack to the atmosphere.

B. Liquid Metal Fueled Reactors

Another homogeneous reactor concept which has received considerable study is the Liquid Metal Fueled Reactor, LMFR. The fuel for such a reactor is a very dilute solution of uranium in bismuth. The bismuth-uranium solution is circulated through an external heat removal system in a manner similar to the aqueous homogeneous reactor. The consequences of a leak in the external system, therefore, are considerable greater than those associated with a heterogeneous reactor. One of the activation products, polonium, is particularly hazardous and toxic (comparable to plutonium).

The waste disposal problems are more properly considered analagous to those of a chemical processing plant, and the actual disposal of wastes will be taken care of in conjunction with the chemical processing.

NATURE OF RADICACTIVE WASTES GENERATED BY PROCESSING SPENT REACTOR FUEL AT THE IDAHO CHEMICAL PROCESSING PLANT

Statement of Phillips Petroleum Company as operator of the ICPP under contract with the Atomic Energy Commission.

To date spent fuel from nuclear reactors has been principally in the category of aluminum-clad, uranium-aluminum alloy, with minor quantities of fuel containing zirconium or stainless steel. Using the technology currently practiced, solid fuel is dissolved in acid, yielding solutions containing uranium, aluminum or other structural material and fission products. This fuel will usually be dissolved between four and six months after discharge from the reactors, but has been more than a year old in some cases.

Usable fissionable material is recovered by a three stage liquid-liquid extraction process leaving a solution containing fission products and dissolved metal. This solution, together with minor amounts of other radioactive process streams, constitutes the principal radioactive liquid waste produced during the process of fuel recovery. The activity of this waste varies over a considerable range depending upon the initial percentage of U-235 in the fuel, the fraction of this uranium which has been consumed in the reactor, the length of irradiation, and the length of the cooling period between fuel discharge and processing. Several methods are available to estimate the radioactivity. These methods yield values which vary considerably. This waste will contain radioactive substances with a total activity equivalent to approximately 1500 curies a gallon at six months after reactor discharge, and approximately one-third that amount at one year. The volume of wastes produced averages approximately 100 gallons per kilogram of residual uranium content of spent aluminum-uranium fuel, and approximately 900 gallons per kilogram of uranium in zirconium-uranium fuel. The relatively high volume

of waste from zirconium fuels results from the low content of uranium in the fuels processed to date, from dilution by chemicals due to the chemistry of the process, and the difficulty of concentrating the waste due to its corrosiveness.

As of January 1, 1959, about 750,000 gallons of first cycle aqueous wastes from six years of intermittent operation on aluminum-uranium fuels are in storage at the plant. Due to heat generation in these wastes as a result of fission product activity, it is necessary to cool the waste volume for a period of up to seven years. Table I gives the approximate composition with respect to the major fission product constituents of a typical aqueous waste from the processing of an MTR-type fuel. Table II shows the chemical composition of first cycle aqueous raffinate exclusive of fission products.

TABLE I

APPROXIMATE FISSION PRODUCT COMPOSITION OF FIRST-CYCLE ALUMINUM-FUEL AQUEOUS RAFFINATE, ONE YEAR AFTER REACTOR DISCHARGE

Element	Total mg/gallon	Radioactive Isotope Mass	Curies/Gallon
Strontium	62	89 + 90	23
Zirconium	200	91 thru 96	35
Yttrium	1	91	24
Niobium	3	95	98
Ruthenium	95	103 + 106	34
Cesium	239	137	12
Cerium	---	141 + 144	162
Promethium	---	147	51
Samarium	27	151	.8
Neptunium	4	237	---
Plutonium	4	239	---
Technetium	27	99	---
Barium	91	---	---
Neodymium	208	---	---
Praseodymium	34	---	---

TABLE II

COMPOSITION OF FIRST-CYCLE AQUEOUS RAFFINATE DUE TO PROCESS CONDITIONS, MOLARITIES

Component	TBP Process	Hexone Process
Aluminum Nitrate	1.6	1.5
Nitric Acid	1.0	3
Sodium Nitrate	0.035	0.004
Ammonium Nitrate	0.01	1.7
Ammonium Hydroxide	0	0.2
Mercuric Nitrate	0.02	0.01
Specific Gravity	1.27	1.3

In addition to the large volume waste from aluminum fuel processing, 30,000 gallons of waste are in storage from processing zirconium fuel. Table III shows the approximate composition of the waste exclusive of fission products. While no absolute data are available on quantities of fission products present, it is thought that the distribution approximates that shown in Table I.

TABLE III

APPROXIMATE COMPOSITION OF FIRST-CYCLE RAFFINATE--ZIRCONIUM FUEL

Ion	Molarity
Aluminum	.62
Zirconium	.47
Hydrogen	1.0
Fluoride	2.6
Nitrate	1.9

In addition to the first cycle aqueous raffinate described in the preceding paragraphs, there exist also 310,000 gallons of combined second and third cycle raffinates. The general composition of this material is similar to the first cycle raffinate except that the fission products present are only a small percentage of the quantity present in the first cycle material. The activity of these wastes is of the order of 10 curies per gallon compared with 1500 curies for first cycle waste.

Other aqueous effluent from the plant, exclusive of sanitary waste, consists of approximately 800 gallons per minute deriving from heat exchangers and condensate from the partial evaporation of liquid wastes prior to storage. This material is discharged to the earth through a disposal well at a level of radioactivity well below the permissable maximum established by the Atomic Energy Commission. Fission product analyses of samples of this waste yield values shown in Table IV.

TABLE IV

FISSION PRODUCT COMPOSITION OF WASTE DISCHARGED TO DISPOSAL WELL

Isotope	Half Life	Per Cent of Total Present Based Upon Gross Beta Activity and Gamma Scan
Strontium-89	54 days	<1
Strontium-90	28 years	$<<1$
Zirconium-95 + Niobium-95	65 days & 37 days	39
Ruthenium-106	1 year	6
Iodine-131	8 days	51
Cerium-144	285 days	4

During the dissolution of spent fuel prior to extraction of residual uranium some radioactive gases are released into the dissolver vapor space. These gases, together with diluent air are either processed for partial recovery of the radioactive gases or are dispersed to the atmosphere under conditions monitored and supervised by the Atomic Energy Commission. Calculations based on chemical analyses of scrubber solution from the Stack Gas Monitor show the radioactive content of discharged gas for a typical month to be that shown in Table V. The fission product krypton is not included.

TABLE V

RADIOACTIVE CONTENT OF STACK-GAS
INTEGRATED TOTAL/MONTH

Iodine-131, Curies	Iodine-132 Curies	Particulate Beta Emitters Other Than Iodine, Curies	Alpha Millicuries
20-30	6-10	40-50	0-10

In a separate process from that designed for uranium recovery, barium-140 and its daughter, lanthanum-140, are recovered from spent fuel. For this purpose, one MTR fuel element is processed in as short a period as practicable after discharge from the reactor. When this is done certain short half-life isotopes are released, notably iodine-131 (8 days). In normally aged fuels, this isotope will have largely decayed to its daughter, stable xenon-131. Fission product iodine-133 is present to a lesser degree plus its daughter, xenon-133. By the time the fuel is processed, fission product iodine-132 will have decayed essentially completely to stable xenon-132. However, tellurium-132 has exhibited a tendency to "plate out" in major off-gas lines and thus provides a significant, although steadily declining, source of iodine-132 for several weeks after the fuel is processed.

In order to limit gross release of iodine and xenon from this process, adsorption and absorption equipment was installed together with gas storage volume to enable retention of evolved gas to allow decay of the majority of short half-life isotopes, prior to release of the gas to the atmosphere through the plant stack. No constituent analyses are available on the total off-gas from the process. The liquid residues remaining after separation of barium and lanthanum contain uranium and are processed in the main plant for recovery of uranium.

The only solid radioactive wastes with which the plant has been concerned to date are equipment, rags, mops and other items which have become contaminated with liquid or gaseous wastes. These materials are handled together with similar wastes from other plant locations and buried after appropriate packaging.

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NATURE OF WASTE FROM RADIOISOTOPE USES

**Presented by Lester R. Rogers, Chief, Radiation Safety Branch
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Magnitude of Isotope Shipments

The Commission's program for distributing reactor-produced radioisotopes (byproduct material) is now over twelve years old, having been initiated on August 2, 1946. During this period about 93,000 radioisotope shipments have been made from the Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee, the Commission's primary radioisotope supplier, to some 4400 institutions throughout the United States for use in the fields of medicine, industry, agriculture and research.* These shipments represent about 437,000 curies of radioactivity.**In addition, it is estimated that the other AEC facilities combined during the same period had shipped to AEC licensees an equivalent amount of radioactivity, the greatest percentage of which consisted of fission products in spent fuel elements and cobalt 60 for applied radiation studies, and polonium 210 and plutonium 239 in neutron sources. Reactor irradiation of special materials and the distribution of small quantities of special radioisotopes such as

* See Table 1.
** See Table 2.

magnesium 28 and iodine 132 accounts for the remaining activity shipped from other AEC installations. Thus the estimated total activity shipped from all AEC facilities to licensees is about 900,000 curies.

For purposes of illustrating the amount of waste produced from the use of isotopes in the United States, the shipment data from ORNL will be used since ORNL is the Commission's largest single supplier of radioisotopes. Of the 437,000 curies of radioactivity shipped from ORNL, about 430,000 curies or 98% had a half-life greater than 30 days. Of this longer half-life material, 429,000 curies consisted of radioisotopes from which sealed sources were fabricated, such as cobalt 60, iridium 192, cesium 137, strontium 90, tritium, and krypton 85. Sealed sources are fabricated and sealed in such a manner that leakage of the radioactive material is not likely to occur under the expected conditions of use, transportation, and storage. These sealed sources all have a half-life of several years and there is no radioactive waste associated with their use; Sealed sources normally become waste only when their radioactivity has decayed to such low levels that they are no longer useful. Sealed sources such as iridium 192 would be subject to waste disposal sooner than those of cesium 137, krypton 85, tritium, or strontium 90 because of its shorter half-life. For example, iridium 192 sources are normally replaced about every three to six

months while a cesium 137 source may be used for several years from the half-life standpoint. Since cobalt 60 is used in sealed sources as a solid piece of metal, there is very little waste associated with its use in sealed source manufacture. There is no waste associated with the fabrication of iridium sealed sources because the sealed source is manufactured prior to irradiation in a reactor. It is estimated that about 80% of the tritium shipped is used in the sealed source form, the remaining 20% being used chiefly in tracer studies. The remaining material in this category having a half-life greater than 30 days which represents about 1,000 curies, consist of radioisotopes used in research and development work involving experimental sealed sources and radioisotope applications which require microcurie or low millicurie amounts of material. The waste from these applications consist primarily of liquids remaining after experimental work, contaminated equipment, paper, and glassware and animal carcasses. A considerable amount of bulk may accumulate from these wastes over a period of time but the specific activity is low.

The remaining 2% of the total activity shipped or approximately 7,000 curies had a half-life less than 30 days and consisted of radioisotopes in the form of liquids, solids or gases which would not be used in sealed sources. The major portion of this activity was radioisotopes of iodine 131, phosphorus 32, and gold 198 which are used chiefly in

medical diagnosis and therapy. It is estimated that a high as 40% of this material becomes waste mostly in the form of excreta from patients. However, because of its short half-life, it does not present a serious disposal problem.

Based on the present types and magnitude of radioisotopes uses, it is estimated that the total amount of waste produced per year from all categories of use is on the order of 100 to 200 curies of material having a half-life greater than 30 days and 400 to 500 curies of material having a half-life less than 30 days. This estimate assumes normal operations. The amount of the longer half-life waste produced may vary widely depending on whether damaged or spent sealed sources are presented for disposal.

Specific Programs

A brief discussion of the types and amounts of waste generated in the various fields of isotope use should indicate the magnitude and type of isotope waste presented for disposal management.***

Medical

Medical radioisotope programs consitute approximately 44% of the total number of radioisotope licensees. The majority of medical users are concerned only with the short-lived radioisotopes, iodine 131, phosphorus 32, and gold 198, for diagnostic and therapeutic use. Diagnostic radioisotope uses are essentially tracer applications and only microcurie amounts of material are used. The therapeutic applications usually involve relatively low millicurie amounts of material per treatment, 100 to 200 millicuries. The waste from therapeutic doses of iodine 131 and phosphorus 32 consist chiefly of the urine of the patients and may represent as much as 60% to 70% of the total amount given the patient. Gold 198 is inroduced into the body cavities or injected into certain

*** See Table 3.

glands or organs for the treatment of cancer. Most of the gold remains in the patient where it undergoes radioactive decay. However, there may occasionally be contamination of bed linens and gowns from drainage. The other wastes produced are associated with the administration of the radioisotopes and the care of the patient. They include microcurie quantities of leftover solutions and solid wastes such as contaminated paper cups and wipes, glassware, syringes, and rubber gloves.

Sealed sources of various shapes and sizes containing cobalt 60 or iridium 192 are used as interstitial transplants for the treatment of cancer. The sources are recovered after treatment, and there is no waste associated with their use.

Medical research programs involve the use of microcurie or low millicurie quantities of such isotopes as calcium 45, tritium, iron 55-59, carbon 14, sulphur 35, sodium 24, and potassium 42 which have both short and long half-lives. In addition to the routine liquid and solid wastes indicated above, animal carcasses, which may contain 25 to 500 microcuries of activity each, may be presented for disposal.

Industrial

Industrial isotope users comprise about 33% of the total licensees. Industrial applications of radioisotopes involve chiefly the use of sealed sources. Firms employing radiography, various types of

gaging, luminous markers, oil-well logging and other sealed source applications account for about 80% of the industrial users. The remaining industrial uses consist chiefly of tracer applications which involve microcurie or low millicurie levels of activity. A few companies which process curie amounts of radioisotopes in fabricating radiography, beta gage and other sealed sources, luminous markers and radioisotope batteries as well as those making isotope-labeled compounds may accumulate millicurie quantities of wastes in the form of solutions and contaminated solid materials having long half-lives and containing the more toxic radioisotopes such as strontium 90.

Sealed Sources:

About 500 firms are licensed to perform radiography with radioisotopes. These firms employ sealed sources of cobalt 60, iridium 192 and cesium 137 having activities ranging from 1 to 100 curies per source.

There are about 450 firms using thickness, density, level, and composition gages containing radioisotopes. A total of about 4,000 of these gages are presently in use. Beta thickness gages usually employ sealed sources of strontium 90, thallium 204 or krypton 85 which normally have activities from 10 to 50 millicuries. Density and level gages usually contain gamma emitting sealed sources of cesium 137 or cobalt 60 in amounts ranging from 50 millicuries to 1 curie. The composition gages such as the hydrogen to carbon ratio gage and the sulfur determination gage contain

small sealed sources of strontium 90 and iron 55 of about 5 to 10 millicuries. Luminous markers for use in aircraft, mines, military vehicles, naval vessels, for standard light sources and low level illumination contain chiefly the isotopes of strontium 90, tritium and krypton 85.

Most of the sealed sources in this category of use have useful lives of several years. The radioactive material contained is in metallic form or as a deposited solid material sealed in metal capsules. When a gaseous radioisotope such as tritium is used in a sealed source, it is often chemically bound to form a solid compound. Krypton 85, however, is usually contained in the gaseous form. As mentioned previously, there is no waste associated with the use of sealed sources. They are only subject to disposal when their radioactivity has decayed to levels which can no longer be used or if they are damaged.

Tracers:

Typical routine industrial tracer applications include interface marking in oil pipelines with antimony 124, wear studies with iron 59, catalyst circulation with zirconium 95, oil-well acidization with iodine 131 or phosphorus 32 and industrial process control with tritium.

There is little or no waste associated with these applications because of the small quantities used and the short half-life of the radioisotope.

Commercial Processors and Suppliers:

Commercial processors and suppliers of radioactive material who prepare special types, forms and compounds of radioisotopes and those firms which fabricate sealed sources are likely to accumulate relatively large quantities of radioactive waste because they may be using bulk, curie amounts of long half-lived alpha-, beta-, and gamma-emitting radioisotopes The waste material produced as a result of these operations consists chiefly of contaminated glassware, handling equipment, paper and residual solutions. The amount of waste may be in the millicurie range and include long half-life radioisotopes.

University and Research Laboratories

University and research laboratory programs are quite diversified and involve a wide variety of sealed sources and other radioisotopes. However, the typical university program does not usually involve large quantities of radioisotopes. In general, the wastes produced are in the form of leftover solutions and contaminated solids such as laboratory equipment, paper wastes, glassware and animal carcasses. The levels of activity of the

wastes are usually low. However, the wastes may include a variety of types and forms of radioisotopes and may include some of the longer half-life isotopes such as strontium 90.

Teletherapy and Irradiators

Teletherapy units for treatment of cancer and irradiators for applied radiation studies represent the largest portion of the cobalt 60 activity shipped. There are presently about 145 teletherapy units installed representing approximately 175,000 curies of cobalt 60 in metallic form. Each unit contains an average of about 1,200 curies of cobalt 60. There are also two medical programs which are using teletherapy units containing cesium 137.

There are presently about 120 irradiators licensed to use more than 100 curies each of cobalt 60 or other radioisotopes. Most irradiators contain kilocurie amounts of cobalt 60. The total amount of cobalt 60 licensed for use in irradiators, but not necessarily installed, is over 1,000,000 curies. Also, about five firms are licensed to use spent fuel elements for irradiation work.

The radioactive material in this category is in the form of sealed sources which have a useful life from 5 to as much as 50 years. There is no waste produced until the radioactive material decays to unusable levels and must be disposed of.

As the activity decays, however, the sources may be used successively for applications requiring lower amounts of radioactivity.

Generally Licensed Items

A number of items containing small amounts of radioisotopes have been generally licensed under Title 10, Part 30 of the Code of Federal Regulations, "Licensing of Byproduct Material." These items may be procured without the filing of applications with the Commission or the issuance of licensing documents. Such devices are manufactured, labeled and tested in accordance with the terms and conditions of a specific license issued to the manufacturers. These applications include the use of nickel 63, cobalt 60, cesium 137, and krypton 85 in electron tubes, polonium 210 in static eliminators, polonium 210 and tritium in ion generators, and strontium 90 in a light meter. The radioisotope in each of these items is sealed in such a manner as to prevent leakage. There is no waste associated with the use of these devices. They are presented for disposal only at the end of their useful life.

Luminous Markers

There are large numbers, perhaps 1,000,000, luminous markers containing low millicurie amounts of such isotopes as strontium 90, promethium 147, thallium 204, krypton 85 and tritium in use, mostly

in the military services. These markers are sealed sources, the level of radioactivity is low, and their use is widely dispersed. At times, however, it is necessary to consider disposal of bulk quantities of these markers when they are collected at a central location from the various components of the military service using them.

Conclusion

The amount of radioactive waste generated through the use of radioisotopes is exceedingly small when compared to the nation's overall nuclear energy program.

As shown, only a small fraction of the radioisotopes distributed are used in applications where significant quantities of waste result during normal use. At the present rate of use of radioisotopes, it is estimated that on the order of 100 to 200 curies of waste with a half-life greater than 30 days and 400 to 500 curies of waste with a half-life less than 30 days are being generated each year.

The quantity of the longer-lived waste may increase significantly in the future as cobalt 60 teletherapy and irradiator sealed sources decay beyond their useful life and are replaced. The amount of waste will be dependent upon whether useful purposes are developed for the spent cobalt 60 teletherapy and irradiator sources.

TABLE I

NUMBER, LOCATION AND TYPE OF LICENSEES

February 10, 1956 - November 30, 1958

States and Territories	Medical Institutes and Physicians	Colleges and Universities	Industrial Firms	Federal and State Laboratories	Foundations and Institutes	Other	Total
Alaska	3	1	0	4	0	0	8
Alabama	33	4	24	13	0	0	74
Arizona	9	3	4	0	0	0	13
Arkansas	21	1	7	4	0	0	33
California	197	10	173	33	3	6	422
Colorado	27	5	14	13	0	4	63
Connecticut	20	5	36	16	2	1	80
Delaware	3	1	11	3	0	0	18
District of Columbia	23	3	3	27	0	1	57
Florida	49	6	12	8	0	2	77
Georgia	27	4	14	7	0	0	52
Hawaii	11	1	2	25	2	0	41
Idaho	4	2	2	0	0	0	8
Illinois	106	18	100	28	5	3	260

NUMBER, LOCATION AND TYPE OF LICENSEES (CONTINUED):

States and Territories	Medical Institutes and Physicians	Colleges and Universities	Industrial Firms	Federal and State Laboratories	Foundations and Institutes	Other	Total
Indiana	41	6	40	6	0	1	94
Iowa	26	5	9	20	0	1	61
Kansas	24	2	10	5	0	0	41
Kentucky	18	3	12	2	0	2	37
Louisiana	27	3	26	4	0	3	63
Maine	5	2	14	5	1	1	28
Maryland	22	4	21	18	0	1	66
Massachusetts	51	27	81	14	6	2	181
Michigan	79	8	45	30	3	2	167
Minnesota	36	6	18	76	0	1	137
Mississippi	8	3	4	1	0	0	16
Missouri	56	8	21	20	1	4	110
Montana	9	2	2	2	0	0	15
Nebraska	15	1	2	1	0	1	20
Nevada	4	1	1	2	0	0	8
New Hampshire	5	3	4	5	1	0	18

NUMBER, LOCATION AND TYPE OF LICENSEES (CONTINUED):

NUMBER, LOCATION AND TYPE OF LICENSEES (CONTINUED):

States and Territories	Medical Institutes and Physicians	Colleges and Universities	Industrial Firms	Federal and State Laboratories	Foundations and Institutes	Other	Total
New Jersey	66	4	112	17	2	2	203
New Mexico	13	3	6	4	0	0	26
New York	280	22	160	58	6	8	534
North Carolina	39	6	12	9	0	1	67
North Dakota	9	2	0	1	0	0	12
Ohio	101	11	71	17	4	2	206
Oklahoma	25	2	31	5	2	1	66
Oregon	18	3	6	5	0	0	32
Panama	0	0	0	0	0	0	0
Pennsylvania	95	16	137	44	2	0	294
Puerto Rico	9	2	0	3	0	0	14
Rhode Island	8	2	12	3	0	0	25
South Carolina	9	2	4	1	0	0	16
South Dakota	11	1	0	9	0	0	21
Tennessee	37	7	20	7	0	2	73
Texas	128	8	90	25	4	3	258
Utah	9	4	4	5	0	1	23

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NUMBER, LOCATION AND TYPE OF LICENSEES (CONTINUED):

States and Territories	Medical Institutes and Physicians	Colleges and Universities	Industrial Firms	Federal and State Laboratories	Foundations and Institutes	Other	Total
Vermont	2	1	2	1	0	0	6
Virginia	34	7	24	12	0	3	80
Washington	24	4	16	10	0	1	55
West Virginia	21	1	10	4	0	0	36
Wisconsin	52	3	32	4	5	1	97
Wyoming	8	1	2	3	0	0	14
TOTAL	1,954	260	1,463	639	49	61	4,424

TABLE 2

BREAKDOWN OF OAK RIDGE NATIONAL LABORATORY SHIPMENTS TO AEC LICENSEES

August 1946 - November 1958

	Curies
TOTAL ACTIVITY OF ISOTOPES SHIPPED	437,000
Half-Life Greater Than 30 Days	430,000
Half-Life Less Than 30 Days	7,000
ISOTOPES USED PRINCIPALLY FOR SEALED SOURCE MANUFACTURE 98%	429,000
Cobalt 60	373,000
Cesium 137	15,000
Iridium 192	22,000
Tritium	15,500
Strontium 90	400
Krypton 85	3,000
NON-SEALED SOURCE MATERIAL 2%	8,000
Medical Isotopes	7,000
Iodine 131	5,300
Phosphorous 32	1,200
Gold 198	400
Chromium 51	40
Others (research and development)	1,000

TABLE 3

TYPE OF WASTE

TYPE OF ISOTOPE PROGRAM	Level		Half-Life		Solid					Liquid	Other
	Low * uc-mc	Medium * mc-c	Short	Leng	Paper	Glass-ware	Sealed Sources	Bed-Linens Gowns	Anim. Car-casses	Resid. Solu-tions	Excreta
MEDICAL	X		X		X	X		X	X	X	X
INDUSTRY											
Radiography		X		X			X				
Gaging		X		X			X				
Tracer	X		X	X	X	X				X	
Research	X	X	X	X	X	X	X			X	
Isotope Processing	X	X	X	X	X	X				X	
UNIVERSITY	X	X	X	X	X	X	X		X	X	X

*uc - Microcuries
mc - Millicuries
c - Curies

Representative HOLIFIELD. At this time we will ask Mr. Herbert Parker, the manager of the Hanford Laboratories, to come to the witness chair.

Mr. Parker, you have been a longtime authority in this field and you have probably one of the biggest jobs of disposal of waste because of the tremendous weapons operations that we have out there.

We are happy to have you here this morning, sir.

I see you have a real extensive statement which is going to be invaluable in our printed hearings. I assume you are going to summarize it.

STATEMENT OF HERBERT PARKER,[3] MANAGER, HANFORD LABORATORIES

Mr. PARKER. Yes.

Mr. Chairman and members of the committee, it is a pleasure to be here and I am tempted to begin by disqualifying myself from the status of expert.

I would specifically for the record disqualify myself from the title of Doctor. My name is H. M. Parker. I am employed by the General Electric Co. at the Hanford Atomic Products Operation in Richland, Wash., as manager of the Hanford Laboratories there.

My remarks are intended to describe only the overall aspects of waste management at that location.

With your permission, Mr. Chairman, I would like to ask one of my associates, Dr. Foster, to manipulate some supporting charts for us.

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[3] Place of birth: Accrington, England. Date of birth: Apr. 13, 1910.
Naturalized in the State of Washington, 1946.
Marital status: Married. Nurmer of dependents: 5.
Education: Manchester University (England); B. Sc. Physics, 1930; M. Sc. Physics, 1931; F. Inst. P., 1937.
Assistant professor radiology, University of Washington, 1952 to date.
Honorary trustee, Northwest Scientific Association (past).
Technical adviser, U.S. Delegation, Peaceful Uses of Atomic Energy, Geneva, 1955.
Janeway lecturer, 1955.
Qualified in Radiological Physics by the American Board of Radiology.
Work history:

1932 to 1938: Physicist, Holt Radium Institute, Manchester, England.
1938 to 1942: Physicist, Tumor Institute, Swedish Hospital, Seattle.
1942 to 1943: Research associate, Metallurgical Laboratories, University of Chicago.
1943 to 1944: Section head, health physics, Clinton Laboratories, Oak Ridge, Tenn.
1944 to 1946: Manager, health physics, E. I. duPont Co., Richland, Wash.
1946 to 1948: Assistant superintendent, medical department (in charge of radiation protection), General Electric Co., Richland.
1948 to 1951: Superintendent, health instruments, General Electric Co., Richland.
1951 to 1956: Director, radiological sciences, General Electric Co., Richland.
1956 to present: Manager, Hanford Laboratories, General Electric Co., Richland.
Professional societies:
Fellow of the Inst. Physics (Great Britain); Fellow, Am. Phys. Soc.; Fellow, A.A.A.S.; Honorary Member, Faculty of Radiologists, Great Britain; Member, Brit. Inst. Radiol.; Member, Am. Radium Soc.; Member, Rad. Soc. N.A.; Member, Am. College of Radiol.; Member, Rad. Research Soc.; Member, Am. Nuclear Soc.; Member, Atomic Industrial Forum; Member, N.Y. Academy of Sciences; Member, Am. Agmt. Assoc.
Committee memberships:
Member, Committee on Units, Standards, and Protection, Am. College of Radiology.
International Committee on Radiation Protection—Chairman, Subcommittee on Isotopes and Waste Disposal (past).
National Research Council—Member, Subcommittee on Radiobiology (past); Member, subcommittee on Rad. Inst. (past).
National Academy of Sciences—Member, Committee on Waste Disposal (Biol. Effects of Radiation Study).
Member, Technical Advisory Committee of the Am. Inst. of Physics to the National Bureau of Standards.
Member, General Electric Reactor Safeguards Council (past).
Member, Executive Committee, N.C.R.P.
Member, Subcommittee on Permissible Dose from External Sources.
Member, Subcommittee on Permissible Internal Dose.
Member, Editorial Advisory Board—Am. Nuclear Society.
Member, AEC Safety and Industrial Health Advisory Board (past).
Member, Committee on Radiation Protection Standards, Atomic Industrial Forum.

Representative HOLIFIELD. Fine.

Mr. PARKER. The Hanford plant is located in the southeastern portion of the State of Washington. It is one of the Nation's major plutonium producing complexes, operated, since 1946, by the General Electric Co. under contract to the Atomic Energy Commission.

Exhibit 1 (p. 184) shows the general location. The area was acquired in 1943 by the United States Government and today consists of a primary zone of 567 square miles.

A secondary zone to the north of about 164 square miles was released as recently as December of 1958 for irrigation and settlement.

The terrain in these parts is typical of arid countries with a rainfall of about 7½ inches per year.

The reservation itself is bounded on the north and west by mountain ranges.

The principal operations of the plant are sketched roughly in exhibit 2 (p. 185). These are the fuel preparation, the irradiation of these prepared fuels in the reactors, the chemical separation of the principal product, plutonium, from these irradiated fuels, and various supporting research and development activities.

Each of these operations generates its own characteristic wastes and their management will be described in sequence.

I will omit that relating to fuel preparation because this is in fact insignificant compared with our other activities. (See pp. 171 and 172 of Mr. Parker's formal statement.)

I will go on to the reactor operations. The fuel elements are subject to nuclear chain reactions for the production of plutonium; highly radioactive fission products are also produced simultaneously.

The process generates heat, and the fuel elements are cooled by the Columbia River water passed through the reactor tubes. The radioactive waste disposal problems at this step in the Hanford process are principally those associated with this cooling water.

During its passage through the reactor, traces of minerals in it are made radioactive as a result of bombardment by neutrons. Small quantities of fission products arise from natural uranium found in the water and from occasional rupture of the fuel elements. The cooling water or reactor effluent thus contains a complex mixture of many different radioisotopes.

The next charts will show for scoping purposes how many are to be dealt with. The first chart (exhibit 3, p. 186) of these will show the number which are neutron activated, those created by the bombardment of the water by neutrons in the reactor.

There are some 350 on this chart.

The next chart (exhibit 4, p. 187) actually lists the principal fission products. If this were an inclusive chart there would be some 250 entries on this one.

The next charte (exhibit 5, p. 188) shows those which we can reasonably measure in the actual reactor effluent water in three categories on this chart on exhibit.

The red ones, [darkest in the black and white reproduction] of which there are 7, are the major contributors to the radioactivity and there are 12 in a minor class and some 43 more listed in trace quantities.

The flow of water through a typical reactor area at our site is shown in exhibit 6 (p. 189).

The amount of the activation products which you recall are the principal contributors to radioactivity depend on the amount of stable isotopes present in the water available to be bombarded. Close control of water quality is required to reduce these quantities to a minimum.

That is largely why there is a large water-treatment plant ahead of the supply to the reactor. Some of the foreign materials stick on the surface of the tubes and on the fuel elements and form a film. Some amount of film helps to reduce corrosion in the tubes, but when this builds up to a certain point it interferes with passage of water and has to be purged or scoured out with powder.

This is an important facet of the waste disposal since this removes accumulated radioactive materials.

In all cases before return of the water to the Columbia River the effluent can be held for any time from 1 to 3 hours in large basins or tanks, called retention basins. This is long enough for the radioactive decay to reduce the total radioactivity by some 50 to 70 percent, as the next chart shows on this falling curve (exhibit 7, p. 190).

Then as the material is returned to the river and passes downstream to Pasco, the nearest point of substantial use, there has been a fairly high decay to something less than 10 percent of the original level of activity as we released the material.

The retention basins serve a separate purpose in that we can use them to intercept any off-standard effluent that may arise from occasional rupture or other defect in the systems and the water can, if need be, be diverted to trenches rather than to prompt return to the river.

The normal effluent as it goes to the river does so through large pipes which are buried in the river bottom. The emerging water is immediately mixed with a large volume of the normal river water.

As shown in the chart before you now (exhibit 8, p. 191), there is some tendency for the reactor effluent to hang together and produce a picture essentially like a smoke plume from a stack conveyed to the water medium in which the radioactive water maintains its primary configuration within the main water channel of the river. This is important, too, for its management.

The Columbia River water is used for drinking water at the city of Pasco and Kennewick serving some 25,000 people there. It is used in summer for irrigation of some local farms and there is now a substantial recreational use of the Columbia River for water skiing, swimming, and boating.

Representative HOLIFIELD. Does the discharge there affect the fish life in the river as far as you know?

Mr. PARKER. A substantial presentation on the work on the fish, Mr. Chairman, will appear later. Perhaps it would suit the committee's pleasure to hear Dr. Foster, who is actually in charge of this project.

I can say, to answer the immediate question, as far as we are aware, there has been no damage to fish from radioactivity release in the Columbia River.

Representative HOLIFIELD. It is of such a low order that there is no danger to the cities of Pasco and Kennewick?

Mr. PARKER. That is correct, sir.

The monitoring of this effluent that we release to the river is an important feature. It begins in the reactor, itself. It is monitored immediately before release and it is separately analyzed.

I will state that most monitoring systems tend to measure the total radiation emerging from the systems and this is inadequate for rational control of the situation in the river which, as we mentioned parenthetically before, does depend on how much of the complex mixture of available radioisotopes is actually present.

This is done both analytically and also by Dr. Foster's operation in an aquatic biology laboratory in which toxicity is observed in live fish at all times.

There are minor radioactive wastes from our operation, some release of gases and aerosols to the air, some retention of solid wastes of various kinds which are placed in shallow trenches well above the water table.

But in summary the disposal of consequence from the reactor system is this rather unique system of returning the Columbia River water to the river and following what happens to it thereafter.

I will pass now to the chemical separations operations. Here after removal of the aluminum jackets the irradiated fuel elements are dissolved in acid and the plutonium and the residual uranium separated from the highly radioactive-fission products and purified by solvent extraction processes as Dr. Culler outlined.

During the dissolving step, the first step, various gases such as radio xenon and vapors such as radio iodine are discharged to the air through high stacks.

The primary stages of the separation processes have created some 52 million gallons of high level radioactive liquid wastes, containing the bulk of the radioactive fission products and these are retained in underground storake tanks.

Later process steps have created some 3 billion gallons of so-called intermediate level wastes which have been admitted to the ground at Hanford under carefully controlled conditions.

In addition, almost 30 billion gallons of cooling water, which have only minor and incidental amounts of radioactive contamination, have been released to surface ponds.

The location of our particular separation facility on an interior plateau takes natural advantage at our site of the high elevation that we gain for the stack release of the gas, the deep dry sediments for the downward percolation of such liquids as we put in the ground and the 7 to 10 miles distance from the Columbia River for possible underground travel of liquids.

Representative HOLIFIELD. On page 5 you say that 52 million gallons of high-level waste is retained in underground tanks. Then you say that there has been 3 billion gallons of intermediate-level wastes which have been admitted to the ground under carefully controlled conditions.

Now, can I assume that those 3 billion gallons are admitted to the earth and are outside of tanks? Are they in tanks?

Mr. PARKER. The 52-million-gallon quantity, Mr. Chairman, is the amount retained in tanks. The 3 billion gallons of so-called intermediate wastes have been introduced to the ground mainly through special structures described in one of these charts, I believe, as cribs.

Representative HOLIFIELD. There is no danger of contamination from the 3 billion gallons in the underground water supply?

Mr. PARKER. The putting of large quantities of material into the ground would be something that in our opinion and from our experience to date at Hanford has been handled with safety.

It would be something that we would maintain should not be contemplated without a very substantial concurrent system of investigation, including laboratory work and field studies.

Representative HOLIFIELD. Do you have a peculiar type of underground water level there which allows you to place this material where it will not get in the underground streams?

Mr. PARKER. We have a fortunate situation on the plateau of having some 200 feet of initially dry sediment which has two properties—first, that as the water goes into it at first it acts, as it were, as a sponge and the water is physically retained. In that way nothing would descend to the lower bed. Secondly, there comes a time when such a sponge becomes saturated and there is a downward passage of the water containing the radioactive materials through it and there certain constituents of the soil remove materials from the water.

This happens to be the radioactive materials. The point not always brought out, Mr. Chairman, is the fact that the radioactivity is inconsequential as far as the soil is concerned. The soil removes particular chemical things which in our case happen to be radioactive.

We are saying then, that the residual drainage of radioactive liquid to the underground water should be eliminated completely if possible and otherwise terminated at some small prescribed level.

Representative HOLIFIELD. How far below the ground is the top of your tanks?

Mr. PARKER. The high level tanks?

Representative HOLIFIELD. The high-level-waste tanks.

Mr. PARKER. These are some 20 feet or so, as I recall, this depth being chosen mainly to shield the people on the ground level from actual radiation from the tanks.[4]

Dr. Foster has a chart of such a tank.

Representative HOLIFIELD. It goes without saying that any of these areas where such material is stored would present a tremendous hazard in case of bombing by enemy attack in the case of the unfortunate landing of a high explosive bomb in that area.

Mr. PARKER. If the bomb should fall in these tanks and should project material from them this would indeed be a problem.

Representative HOSMER. What would it amount to in relation to the problem of radioactivity created by the bomb?

Mr. PARKER. That is an extremely difficult question to answer in categorical terms.

Representative HOLIFIELD. The answer to that is that with 52 million gallons of high level radioactive liquid wastes, you would have a much greater contamination and more permanent contamination that you would from a bomb that was exploded, even though it might be of the megaton variety.

Mr. PARKER. Assuming, sir, that the bomb was not created specifically to cause like contamination, then it would be true, this would

[4] The value of 20 feet is representative for the depth of the tank liner. The actual tank has a dome top whose depth below the ground varies from about 6 feet at the center to 15 feet at the edge.

be a significant problem from such long-lived materials as strontium 90 and cesium 137 in this 50 million gallons of liquid.

The material that we presented to the committee, Mr. Chairman, does not include reference to actions of this type. It would be relevant, if one considers this, to consider it also on the reactor structure. We interpreted our instructions to omit this.

Representative HOLIFIELD. The same hazard exists with any large-sized reactors. If any of these large-sized reactors loaded with fuel elements should be the target of enemy attack you would have tremendous contamination of all the surrounding communities from an explosion of a reactor itself whether it was done by enemy bombs or by other types of explosives. It is one of the hazards you have to live with in this age.

Mr. PARKER. Yes, sir. Our local reviews of this could be submitted to the committee if it is your pleasure.

If we may go on to a brief mention of the separate kinds of wastes that arise from our chemical separations, let us look first at the material that goes to the atmosphere. Most of the gaseous and vapor material is released in the first step by dissolving the fuel elements, but some comes through a later step. The exhibit now before us (exhibit 9, p. 192), a rather complex representation of tanks and the like, we may take for our present purpose to show only that we must be careful in the first steps of our process to collect gas and vapors that come from any process vessel and funnel them through a cleaning mechanism. The kinds used are a combination of chemical systems, the so-called silver reactor for removal of iodine at the beginning and at the other extreme a final taker-outer of radioactive particles which is this large box structure, which is a sand filter.

All the waste in the chemical-processing building eventually goes through such a structure so that none is returned to the atmosphere without removal of a substantial fraction of the radioactive particles it may have acquired in our manipulations.

The plutonium-purification facilities are separate from the ones intended to be depicted in this chart (exhibit 10, p. 193) and they also have had filters, in this case from the very beginning of our operation. They have been recently improved by finding a variety which is fire resistant to improve the safety of that method of control.

Materials that leave the stack are monitored continuously by special sampling devices so that we have a current record of the release of such materials as radioiodine.

It turns out that in our system the radioiodine is one of the controlling factors which has to be followed very carefully. This arose because the hazard is an indirect one. It is not related to the matter of men breathing radioiodine but relates to the indirect process of the material depositing on vegetation and, therefore, to a degree accumulating. This vegetation may then be eaten by grazing animals or produce or, ultimately, the animals eaten by man.

This, I would say is characteristic of the real problems in waste disposal, that they will end up in complex chains, each of which has to be followed.

Mr. RAMEY. Is this the type of material that is concerned with the so-called Wahluke Slope problem?

Mr. PARKER. The consideration on the Wahluke Slope problem, Mr. Ramey, concerns itself largely with that part which we have omitted from the presentation today, that is: what would happen in the event there was a substantial accident to an operating reactor and dispersal of the total contents, in which case the iodine would most likely be not the most consequential problem.

Going to the storage of the high-level-liquid wastes, you will recall that the bulk of the fission products present in our fuel elements will end up in this system.

Exhibit 10 shows the kind of vessel in which these wastes are maintained. They are actually described as reinforced concrete tanks with a steel liner.

Representative HOLIFIELD. What is the thickness of the concrete around the steel?

Mr. PARKER. It is a matter of some feet. What that dimension is I do not recall specifically.[5] The purpose of the concrete is to add strength to this structure. The isolating membrane between the radioactive materials and the environment, which is the essential thing to maintain intact, is in fact, the steel liner.

Representative HOLIFIELD. The concrete acts as a neutron buffer, too; does it not?

Mr. PARKER. At this location, sir, the radiation is almost completely so-called gamma radiation and beta radiation. The neutron factor would be essentially insignificant.[6]

Representative HOLIFIELD. Is there any strontium and cesium in it?

Mr. PARKER. Strontium and cesium are in this vessel and other stronger emitters of radiation. But the vessel is primarily for mechanical integrity, not for radiation shielding, since we are going to achieve this by burying the tank.

We should stipulate that the use of tanks of this nature is a waste storage rather than waste disposal, and that no environmental hazard will exist as long as the tanks maintain their integrity.

We have to report that the life of the tanks is not yet known. None has as yet failed and the life is estimated to be at least several decades.

Moreover, a modest leak in a tank would not create a major hazard because the tanks are located over the available 200 feet of dry sediments.

Representative PRICE. What do you believe the life of the tank to be?

Mr. PARKER. I will answer that question by saying that for a longer time than any operation heretofore contemplated by man, these wastes will have to remain isolated from the environment and until the time we create a better way the isolation will be in tanks of this character.

This does not mean it will have to be in this particular tank. In other words, if the tanks we have turn out to have a life of 50 years, it will be very simple to be prepared at the right time with an alternate set of tanks and pump the liquids into new tanks.

We have extensively moved the liquid from one tank to another and are persuaded that we can do this operation with perfect safety.

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[5] The concrete wall is 2 feet thick.

[6] The reference here is to the fact that if there are alpha-particle emitters in the waste and certain substances, such as fluorine, of low atomic number, there can be a nuclear reaction that releases neutrons. As a practical matter, in the high level fission product wastes, this is insignificant.

Representative PRICE. How many years are expected for the storage tanks that are already existing?

Mr. PARKER. Tanks in general, sir, rather than tanks supplied for this purpose?

Representative PRICE. Supplied for this purpose for the storage of liquid wastes.

Mr. PARKER. There will be a difference here because those wishing to take the position that tank storage is not a very desirable thing would persuade you that there were materials in our liquids which would cause more rapid deterioration than in other industrial applications. Without that these tanks should well have lasted from 100 to 200 years.

Representative HOLIFIELD. We are facing the inevitability of pumping this material out of these tanks and into new tanks to prevent any leakage because of the deterioration or the corrosion of steel?

Mr. PARKER. That is inevitable with one reservation—that we expect in fact, before the time to do this pumping, to demonstrate that there is a superior method of packaging materials in dry form. This may prove to be the reasonable and economic solution.

If not, we will say that ultimately these wastes will in fact be moved from one tank to another set of tanks.

Representative HOLIFIELD. Are experiments going on at this time in the way of reducing this from liquid to solid?

Mr. PARKER. Experiments are going on in several locations in the country, including our own efforts, which are currently proposed to be accelerated.

Representative HOLIFIELD. We have a witness who will testify in detail on that later, I believe.

Representative PRICE. Is the situation serious enough to cause you to transfer from one tank to another?

Mr. PARKER. No; we have not. We have had what might be described as suspicious occurrences in these tanks. With a tank as large as this it is difficult to create suitable devices, since there is no access to the device, to measure the liquid level with precision. There have been times when there have been apparent oscillations in these liquid levels and substantial investigations as to whether material has in fact leaked have been made.

If we may answer the question in that framework, we have never detected a leak from any of these tanks, so that we are in turn persuaded that none has ever leaked.

Representative PRICE. Are there any other places where they have leaked?

Mr. PARKER. The experience that I have, sir, and on which I am reporting is confined to the Hanford location. I am not aware of any at other locations, but would not feel qualified to make a definitive comment.

One additional factor that we did not mention is that a considerable part of the radioactivity that is retained in the tanks becomes assocated with a heavy sludge that falls to the bottom of the tanks and one suspects in the event of rupture of the tank this would have relatively little inclination to travel underground. In more recent years a program of two things, so-called self-concentration and

scavenging of waste followed by ground disposal of the purified part, have enabled us to conserve our storage space. Self-concentration is simply the boiling away of our more vigorous wastes by the heat produced by their own radioactivity.

Scavenging is a cleaning up of the more insidious of the elements in these tanks, items such as strontium 90 and cesium 137; removing these by a suitable chemical process and then, as it were, degrading the large volume of residual waste to the class of what we call intermediate level waste and putting that in the ground.

Representative HOLIFIELD. Does this method of self-concentration decrease the life of the tanks because of the increase in corrosion?

Mr. PARKER. Increase the life of the tank?

Representative HOLIFIELD. Decrease the life of the tank because of the increase in the factor of corrosion. You have to use heat to do this as I understand it.

Mr. PARKER. It would be reasonable that under the boiling conditions the tanks would require closer examination for this factor than heretofore.

The number of years boiling has taken place is not sufficient to characterize this, but it would be expected to be in this direction. We maintain as I mentioned, an inventory of the contents of each tank and follow within experimental limitations the liquid levels.

In addition, we have dry wells drilled adjacent to the tanks and in some cases as far down as the water table as an additional means of detecting and evaluating leaks should any occur.

Passing from the high-level wastes to those which we will call intermediate and low level, these are the ones that have been sent into the ground through a variety of structures, some known as cribs, in which the main volume of intermediate wastes go to the ground, the water ending up in the underground water table (exhibit 11, p. 194) shows these disposal methods.

The radioactive content is retained in the earth materials. The item shown on the sketch (exhibit 11) shows a trench for smaller volumes of waste which tend to be not compatible with our main volumes, so we put them out in a corner and let them soak in their own particular vertical earth column.

This method of disposal requires—and we would consider this mandatory—that each major type of waste be evaluated in the laboratory for its behavior in soil before disposal operations are started, and that after the disposal is underway their possible pattern be followed underground through a network of wells.

More extensive consideration of this part will appear in the research and development portion. The low-level waste itself, the 30 billion gallons, mostly cooling water, would normally have no radioactive content but since they are in contact with process vessels and there is a possibility of occasional leaks, they are looked at in the event there is some transfer of material.

This enormous volume, however, is disposed to the ground through these so-called swamps or surface ponds [pointing again to exhibit 11]. The disposal of this very large amount of water has, in fact, demonstratively modified the situation that existed at Hanford before.

It has created mounds of water standing on the water table, in

one case as high as 70 feet. Such mounds will appreciably modify the previous hydraulic gradients and it is important that the possible accelerations of movements be followed in our planning.

I will omit reference to our research and development operation which create waste, again trivial in comparison with the main quantities.

To cover very briefly the total magnitude of our disposal efforts at Hanford, the capital cost of the relevant facilities are said to be some $114 million.

The related operating and maintenance cost for waste disposal is some $2 million, and for current research and development activity is some $1 million, annually.

In addition to the numerous measurements which are made on the waste prior to or at the time of release, we at Hanford conduct an extensive program of environmental monitoring and the next chart (exhibit 12, p. 195) simply shows the general size of this kind of program in terms of various classes of samples.

The total manpower that goes into the waste-disposal operation, environmental sampling, and evaluation of our efforts, is about 155 men with another 100 men engaged in various research and development phases.

Representative HOLIFIELD. Does the $2 million include the cost of the environmental monitoring program?

Mr. PARKER. We submitted here a condensed evaluation of the cost. We are in a position to make a more detailed one but do this with reservation, Mr. Chairman, because the nature of our work is such that to separate cleanly that which is purely associated with waste disposal and that which is a clear part of the process, is arbitrary and, it is our opinion, would possibly be misleading rather than constructive. These numbers can be made available if it is your wish.

A note on the bases of protection. At Hanford we have continuously had a separate component prepare local standards and evaluate the plant's performance against the standards, a desirable check and balance system.

Our local standards are not less conservative than those prescribed by the Atomic Energy Commission and are broadly responsive to those promulgated by the standard qualified sources as the National Committee on Radiation Protection and Measurement and the corresponding International Commission on Radiological Protection.

Particularly in the matter concerned with release of materials to the environment the posture has been conservative. Due to our local research, we have limits in many cases substantially more restrictive than would be required by literal following of conventional sources. One good illustration of that is the matter of our release of radioiodine from stacks which is actually some 6,000-fold more restrictive than that derived from contemporary handbooks.

Representative HOLIFIELD. This would indicate an extra degree of caution on your part or a degree of laxity on the contemporary handbook?

Mr. PARKER. It is a matter of terms of reference, I would say, Mr. Holifield, in that the contemporary handbooks refer to the breathing of material from the air and all observers would be in agreement as to the permissible quantities for that.

In the Hanford situation, and one discovered and developed by our local researches, the real problem developing with the release of large quantities of iodine is that this deposits on the vegetation. It accumulates and is eaten in those cases by sheep. We have an experimental farm and we are following the real effects in an extensive research program.

Representative HOLIFIELD. You are to be commended for your caution in this field.

Mr. PARKER. Thank you, sir. We say in the text that research leading to one specific permissible limit of this type may well cost in excess of $1 million, as they have in this case.

We consider a significant feature of our work at Hanford has been the effective integration of purpose between operation and research to modify such standards and to change the corresponding practices used to maintain an assured position of safety in waste disposal.

It is for this very reason of integration between our operation and research that we have some difficulty in sorting out for submission what is clearly operation and what is clearly research and we would like to reccommend to the committee, Mr. Chairman, that our later submissions under "Section IV. Research and Development," be considered along with the waste management as here today.

Very briefly, the long-term exposure of personnel within our control zones is to be limited to 5 rem per year in accordance with the current national view of this, and outside the controlled zones currently to an annual limit of 0.5 rem per year or one-tenth of the occupational limit.

Now, a key feature of the disposal is that any individual may receive radiation exposure from diverse sources—this, in fact, is the crux of the problem, to end up determining the integrated dose to all individuals who may be affected by complex systems.

The next three charts (exhibits 13, 14, and 15, pp. 196–198) show very briefly the principal sources of exposure that are our particular problems at Hanford, arising first, from the cooling water going to the Columbia River, its hazards when used as drinking water, feasible hazards from external radiation, and a battery of circumstances arising from possible food supplies, by aquatic plants, fish, water fowl, irradiated crops.

Representative HOLIFIELD. I was interested in your statement on page 13:

> The deposition of isotopes on local vegetation from fall-out exceeds that which originates from the Hanford plant.

This is interesting because it does show you have done a pretty good job of monitoring your plant operation.

Mr. PARKER. We would consider it a defect in the system to engage in release, in the present state of knowledge, without, as you have said, currently maintaining a program of actually investigating what happens.

If we look for a moment at two corresponding charts—in your copy, gentlemen, the next two charts were inadvertently interchanged—we will take the atmosphere next (exhibit 14, atmosphere) the release of iodine or of particles, leads to breathing hazards, possible hazards

from external radiation from clouds, and mainly from our experience locally, hazard from the deposition of materials, leading to crop contamination, uptake in crops, uptake in animals, using the milk, eggs and meat of those animals, water supplies, and the transfer to man accordingly.

The third channel (exhibit 15, ground disposal) is that of the low- and intermediate-level fission product waste to the ground. I will reiterate that in our system at the present time there has been no hazard of any kind from this release to the ground because the material has not returned to the environment.

So we are sketching here the channels in which this material may later appear. If not buried, there can be hazards at the disposal site due to both external radiation and more insidiously due to the wind pick-up of dry materials, say from ponds that have been used to release material, and the various chains in the uptake of plants and animals.

Once the material is safely underground the use of wells has to be watched. This cannot be done in land that is not controlled for this purpose. Deep rooted plants may bring the material to the surface and ultimately there may be travel of some quantities to the river and hence a recycle into the chart that we showed for river contamination.

That concludes what I propose to say today.

Representative HOLIFIELD. Mr. Parker, the experience that you have had there in Hanford is about one of the most complete in the storage of a great mass of this material and certainly, you have had a very fine record of no large incidents that have been a hazard to the population.

You have done a careful job in handling this very deadly material. Thank you very much.

(The formal statement of Dr. Parker follows:)

RADIOACTIVE WASTE MANAGEMENT OPERATIONS AT THE HANFORD PLANT

Presented by H. M. Parker, Manager, Hanford Laboratories
Hanford Atomic Products Operation
General Electric Company

My name is H. M. Parker; I am employed by the General Electric Company at the Hanford Atomic Products Operation in Richland, Washington, as manager of the Hanford Laboratories. My remarks are intended to describe the over-all aspects of waste management at Hanford.

The Hanford plant is located in the southeastern portion of the State of Washington. It is one of the nation's major plutonium-producing complexes, operated, since 1946, by the General Electric Company under contract to the Atomic Energy Commission. The general location of the area is shown in Exhibit 1. The area was acquired in 1943 by the U. S. Government, and now consists of a primary zone of 567 square miles. A secondary area of about 164 square miles was released in December of 1958 for irrigation and settlement. The terrain in this part of the state is typical of arid countries with a rainfall of about seven and one-half inches per year. The reservation itself is bounded on the north and west by mountain ranges.

The principal operations of the plant are, as shown in Exhibit 2, fuel preparation, irradiation of the fuel elements in the reactors, chemical separation of the product plutonium, and research and development. Each of these operations generates radioactive wastes and their management will be described in sequence.

1. Fuel Preparation

In this operation, where uranium-metal fuel elements are prepared and jacketed in aluminum containers, radioactive waste is limited to small quantities of unirradiated uranium. These wastes, together with other liquids which contain non-radioactive chemicals, are released

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to ponds near the Columbia River. In comparison with the uranium found naturally in the river, the quantity which seeps in from the pond is so small that it can be detected for only about a mile downstream. The problems of radioactive waste disposal from this operation are insignificant compared with those of later processes.

2. Reactor Operations

The fuel elements are subjected to nuclear chain reactions for the production of plutonium; highly radioactive fission products are also produced simultaneously. The process generates heat and the fuel elements are cooled by Columbia River water passed through the reactor tubes. The reactor locations are dictated by the combined needs of access to river water, and relative isolation from populous areas. The latter feature is required for protection only in the extremely improbable event of a major operational accident.

The radioactive waste disposal problems at this step in the Hanford process are principally those associated with the cooling water. During its passage through the reactor, traces of minerals in the cooling water are made radioactive as a result of bombardment by neutrons. Small quantities of fission products arise from natural uranium in the water and from occasional ruptures of fuel elements. The cooling water, or effluent, which leaves the reactor thus contains a complex mixture of many different radioisotopes.

Exhibit 3 shows all the possible neutron activated isotopes and Exhibit 4 shows the principal fission products. Of all these, Exhibit 5 shows the ones that we can actually measure in the reactor effluent water.

The flow of water through a typical reactor area is shown in Exhibit 6. The amount of activation products in the effluent depends on the amount of stable isotopes present in the water. Close control of water quality in the water treatment plants holds this amount down.

Some foreign materials inevitably deposit on the surfaces of the process tubes and fuel elements to form a film. Some film helps to reduce corrosion but when it builds up to the point where there is interference with the coolant flow and heat transfer, it is removed by "purging" with finely powdered diatomaceous earth.

Prior to release to the Columbia River, the effluent can be retained for one to three hours in large basins or tanks. This is long enough for radioactive decay to reduce the gross radioactivity by 50 to 70 per cent, as shown in Exhibit 7. By the time the effluent has traveled to the vicinity of Pasco, the first point of substantial use, radioactive decay has further reduced the gross activity to less than 10 per cent of the level when it entered the basins. The retention basins (Exhibit 6) also serve another purpose in that they intercept off-standard effluent which has an unusually high radioactive content. Such situations arise from the occasional ruptures of fuel elements and from purges. In some cases, the abnormally contaminated effluent water is discharged to trenches along the river bank, where it seeps into the ground.

The effluent enters the river through large pipes buried in the river bottom, and is immediately mixed with a large volume of water flowing in or near the main channel. As shown in Exhibit 8, however, there is a tendency for high concentrations of the effluent to remain in midstream.

Columbia River water is used for municipal supplies by the cities of Pasco and Kennewick which serve approximately 25,000 peopel, and for process water in a few industrial plants. During the summer months it is also used to irrigate a few farms in the vicinity of Ringold and up to about 5,000 acres immediately northwest of Pasco. At this time, there is no major navigation on the Columbia River above Pasco, but recreational boating, including water skiing and swimming, is very popular in the Tri-City Area. Private boats are permitted as far upriver as the boundary of the reservation and fisherman have access to the east shore as far upriver as Ringold.

Monitoring of the effluent is an important aspect in the management of the reactor waste and is accomplished at several different points. The first indication of the radioactive composition of the effluent is obtained from samples which originate from various parts of each reactor. The signals from instruments which scan the samples usually provide the basis for shutting down the reactor when there is a rupture of a fuel element. The effluent is next monitored at the time it leaves the retention basins. While these instruments are valuable as early warning systems, they are not capable of distinguishing between individual isotopes. The concentrations of the individual isotopes or groups of isotopes which might contribute significantly to radiation exposure in the environs of the plant are determined routinely in analytical laboratories. As an added precaution to assure that reactor operations do not affect fish, a continuous sample of the effluent being discharged to the river is pumped to an aquatic biology laboratory and its toxicity tested with live fish.

A minor source of radioactive waste from the reactor operations, shown on Exhibit 2, is the gases and aerosols. Air which is used for ventilation picks up contaminated dust from portions of the building which are normally unoccupied, and gas which leaks from the reactor chassis. The release of these materials through high stacks to the atmosphere has produced no distinguishable contamination levels in the air or on the ground outside of the restricted area.

Solid wastes, such as contaminated paper, boards, worn out tools, construction items, and aluminum spacers used to center the fuel charge, are buried in shallow trenches in the reactor areas. Similar solid wastes are isolated from the enrivonment since care is taken that the bottoms of the trenches are above the water table and the rainfall is low enough so that percolation through the deposits is practically non-existent.

3. Chemical Separations Operations

After removal of the aluminum jackets, the irradiated fuel elements are dissolved in acid. The plutonium and residual uranium are then separated from the highly radioactive fission products and purified by solvent extraction processes. Finally, the plutonium and uranium are processed separately to the required forms.

During the dissolving step, various gases such as radio-xenon, and vapors such as radio-iodine are discharged to the air through high stacks. The primary stages of the separation processes have created some 52,000,000 gallons of high-level radioactive liquid waste, which is retained in underground storage tanks. The later process steps have created some 3,000,000,000 gallons of intermediate-level wastes which have been admitted to the ground under carefully controlled conditions. In addition, almost 30,000,000,000 gallons of cooling water, with only minor and incidental amounts of radioactive contamination, have been released to surface ponds.

The location of the separations facilities on an interior plateau takes natural advantage of the high elevation for the stack release of gases, the deep dry sediments for the downward percolation of liquids, and the seven to ten miles distance to the river for underground travel of liquids.

Although the radioactivity in all of the types of waste described above originates principally from the fission products, each type of waste requires quite different management methods and a brief description of these follows:

a. Release of radioactive material to the atmosphere (Exhibit 9)

While the dissolving of the fuel elements is the principal source of radioactive gases and vapors, more of the same materials are released in the later processing steps. In working practice it is the radio-iodine which predominates as the material requiring

control. Radio-ruthenium has also been present, since at one time some of it was volatilized intentionally at one stage of the process.

The quantity of radio-iodine liberated to the atmosphere is controlled by a combination of aging of the fuel elements prior to processing, which allows radioactive decay of the short-lived I^{131}, and the retention of most of the volatilized iodine in silver reactors. These units contain fixed beds of a porous packing material on which silver nitrate is deposited. The iodine combines with the silver at about 200° C, and is retained on the bed. Periodically the bed is regenerated and the liquid waste which contains the radio-iodine is sent to the underground storage tanks. In 1957, facilities were installed which absorb acid-producing vapors; these absorbers also remove iodine and thus serve to back up the silver reactors.

Most of the ruthenium which is volatilized is removed in caustic scrubbers. Some of it may change in chemical state, however, to form minute particles and these are effectively retained in filters before the air is released to the stacks.

The ventilation air of the process buildings is also vented to high stacks because it may, from time to time, contain minor releases from process vessels. It is first passed through large filter beds of graded layers of sand or mats of glass fiber which remove virtually all of the entrained radioactive particles.

The plutonium purification facilities have always been equipped with filters which remove radioactive materials from process gases before they are released, and no significant contamination of the atmosphere from these facilities has yet been experienced. The original filtering system utilized paper filters developed by the Chemical Warfare Service. In 1957, these units were replaced by fire-resistant units developed especially for this purpose.

The amount of radioactive material which leaves the stacks is monitored continuously from samples passed through moving filter strips or through solutions which retain the radio-iodine. These measurements, together with others made at various points in the process, provide a continuous record of the release.

Close control of the quantity of radio-iodine released to the atmosphere is maintained not only to limit the amount in air, which is breathed, but more restrictively, to limit the amount which deposits on vegetation eaten by grazing animals and produce eaten by man.

b. Storage of high-level liquid wastes

The bulk of the fission products which were present in the irradiated fuel elements are retained in the solutions as the plutonium and uranium are removed during the primary stages of the separation process. These high-level wastes are stored as alkaline slurried in underground tanks such as shown in Exhibit 10.

The tanks are constructed of reinforced concrete and they are lined with steel plate. Each has a capacity of 500,000 to 1,000,000 gallons. Placement of the tanks well below ground serves the dual purpose of shielding personnel from their radiation, and of protecting the tanks themselves.

The use of such tanks is generally considered to be waste storage rather than waste disposal and no environmental hazard exists as long as the tanks maintain their integrity. The life of the tanks is not yet known, since none has ever failed, but it is estimated to be at least several decades. Moreover, a modest leak would not create a significant hazard since the tanks are located some 200 feet above the water table and the dry sediments beneath the tanks have a large

capacity for retention of liquid waste. Furthermore, a considerable part of the radioactivity becomes associated with a heavy sludge that should have little inclination to travel underground.

In recent years programs of self-concentration and scavenging of wastes followed by ground disposal of the purified supernate have recovered storage space so that additional tank construction has been postponed. Self-concentration is accomplished when a sufficient amount of heat is released by the radioactive decay of isotopes to cause boiling of the liquid in the tanks. In old, non-boiling wastes, the volume can be reduced by separating out the long-lived isotopes which have not been eliminated by radioactive decay and returning them to the tanks in a more concentrated form. The use of these methods for the recovery of space in the tanks sets a reasonable limit to the ultimate volume which must be stored.

An inventory is maintained of the contents of each tank and the liquid level is watched carefully. Furthermore, dry wells have been drilled adjacent to the tanks and, in some cases, down to the water table as an additional means of detecting and evaluating a leak, should this occur.

c. Disposal of intermediate-and low-level wastes to the ground

Liquids which have an intermediate concentration of radioactive materials include waste streams from the later decontamination steps of the separations process, spent solutions used to absorb or scrub the gases released to the atmosphere, condensed vapors from the self-boiling tanks, various other condensates from process equipment, and the aged solutions from which the long-lived isotopes have been scavenged.

The intermediate-level wastes enter the ground by seepage from structures known as cribs and trenches which are shown schematically in Exhibit 11. The cribs are located underground to minimize radiation

levels and contamination at the ground surface. Open trenches are used for isolated batches of waste that would cause complications at other disposal sites. They are refilled immediately after a single disposal operation to prevent the spread of surface contamination by winds.

Because of the favorable geological and hydrological conditions at the separations area and the capacity of the soil to adsorb the isotopes, it is possible to retain the vast majority of the radioactive materials in the thick layer of sediments. Thus, the wastes are essentially "stored" in the ground and the water percolating to the water table is purified. Were the radioactive materials to enter the ground water, additional decay time and retention capacity would be available during the relatively slow movement of the ground water to the Columbia River where copious dilution would be available.

Each major type of waste is evaluated in the laboratory for its behavior in the soil before disposal operations are started. After disposal is under way, a network of wells is used to follow the travel of isotopes through the ground and any transfer to the groundwater. In this program, any process waste stream which would not be properly retained in the soil is treated as a high-level waste and stored in tanks.

The so-called low-level wastes consist, for the most part, of cooling water and utility steam condensates which are ordinarily non-radioactive. However, since these waters may occasionally be slightly contaminated by leaks from process vessels, they are considered in the category with other low-level wastes.

This waste, which is essentially uncontaminated, is discharged to surface ponds or swamps from which it seeps through the soils to the water table some 200 or more feet below. The large volume released has built an artificial mound of water on top of the original

water table under each of the two separations areas. The western mound, in fine sand and clay, stands about 70 feet high; the eastern mound, in more porous sand and gravel is spread out and stands about 20 feet high. These mounds substantially modify the natural hydraulic gradients and so may alter the paths of any radioactive material which reaches the groundwater.

4. Research and Development Operations

The majority of the research and development facilities are located in the Laboratory Area, a few miles from Richland. There are also important experimental facilities located in the reactor and separations areas, however. These operations create radioactive wastes which are similar in kind to the wastes of the main process, but the quantity of radioactivity is, of course, much less. Although their disposal requires care, they present trivial problems in comparison with the process wastes with which they are usually combined.

Magnitude of Waste Disposal Effort

The capital costs of all Hanford facilities or systems involved in the collection, handling, processing, monitoring, and control of gaseous, liquid, and solid waste effluents are approximately $114,000,000. The related annual operating and maintenance costs are about $2,000,000, while research and development activities concerned with waste disposal problems are conducted at a level of $1,000,000 annually.

In addition to the numerous measurements which are made on the waste prior to, or at the time of, release, Hanford conducts an extensive program of environmental monitoring in order to evaluate the radiation exposure in the environs of the plants. Exhibit 12 enumerates the principal samples involved in this environmental or field survey program.

The total man-power involved in waste disposal operations, environmental sampling, and evaluation is about 155 man years per year. Related research and development activities require about 100 man years per year.

Bases of Radiation Protection at Hanford

Working radiation protection standards and practices at Hanford are established, and modified as appropriate, by a separate Company component, which also evaluates the actual performance against these standards. Responsibility for operation within such limits always belongs to the respective managers of production operations, thus providing a desirable check and balance system.

The local standards are not less conservative than those prescribed by the Atomic Energy Commission and are broadly responsive to those promulgated by other qualified sources such as the National Committee on Radiation Protection and Measurement, and the International Commission on Radiological Protection. Particularly in matters concerned with releases of radioactive material to the environment, the posture has been conservative. Due to local research, limits substantially more restrictive than those to be inferred from conventional sources have frequently been developed. As an example, the permissible release of radio-iodine from stacks is limited at a level 6000 fold more restrictive than that derived from the contemporary handbooks. This is required because the relevant radiation hazard for this particular isotope depends on its deposition on vegetation and its potential consumption by range animals. The handbook limit is based on the permissible inhalation of the material by man.

Research leading to a specific permissible limit of this type may cost in excess of one million dollars, and additional operation control equipment to achieve the lower limit may cost several millions of dollars. Such expenditures are a necessary part of a comprehensive waste management program. A significant feature of the Hanford operation has been the effective integration of purpose between operations and research to modify both standards and practices to maintain an assured position of safety in waste disposal. For this very reason, it is difficult for us to separate practice from research and development

in reporting here. The later submissions of my associates under Section IV - Research and Development should be considered jointly with today's presentations.

Waste disposal policy is designed to limit the exposures of individuals and other significant life forms to the lowest practicable level. In essence, the long term exposure of personnel within the controlled zones is limited to a maximum of 5 rem per year. Outside the controlled zones, an annual limit of 0.5 rem per year is considered appropriate. An individual may receive radiation exposure from diverse sources, some of which materializes as a result of the accumulation and transfer of radioactive materials through complex systems. The integration and precise evaluation of these exposures is quite complex and such interpretations will be included in written material submitted to the Joint Committee.

Exhibits 13, 14, and 15 show the sources of exposure and complex chains that we consider as requiring investigations before a given waste is judged suitable for release to the environment. The examples relate, respectively to the Hanford release of:

a. Cooling water to the Columbia River.

b. Process gases, vapors, and particles to the atmosphere.

and c. Low and intermediate level fission-product wastes to the ground.

Hanford waste management follows a so-called point-of-exposure criterion, which involves control of discharge from all sources so that no permissible limit is exceeded at any point of exposure, which may be remote, both in distance and time, from the point of discharge. It also recognizes and incorporates the effects of dilution, radioactive decay, and reconcentration of radioisotopes in biological systems.

In brief summary, the actual radiation exposure to persons in the environs of the Hanford plant is estimated to be between 3 per cent and 15 per cent of the currently accepted limits. For exceptional cases,

where unusual amounts of local fish and leafy vegetables are eaten, the exposure may reach as much as 40 to 60 per cent of the appropriate limits. These values include not only the contributions of radioactive materials from the reactor effluent and the separations plant stack gases but also from the worldwide fallout pattern. The deposition of isotopes on local vegetation from fallout exceeds that which originates from the Hanford plant.

Relationships with Other Organizations

Various Federal and State agencies have contributed to the evaluation of Hanford waste disposal practices. They have reviewed existing research programs, suggested additional avenues for study, and worked cooperatively on site, in some cases. As an example, the Hanford Operations Office, AEC established, in 1949, the Columbia River Advisory Group, an informal body comprised of pollution experts of the States of Washington and Oregon, with a chairman representing the U. S. Public Health Service. This relationship has materially assisted both the Commission and the General Electric Company in their discharge of public responsibility. The understanding of Hanford's waste disposal practices thus acquired by responsible officials is believed to have had reciprocal value to the States.

EXHIBIT 1

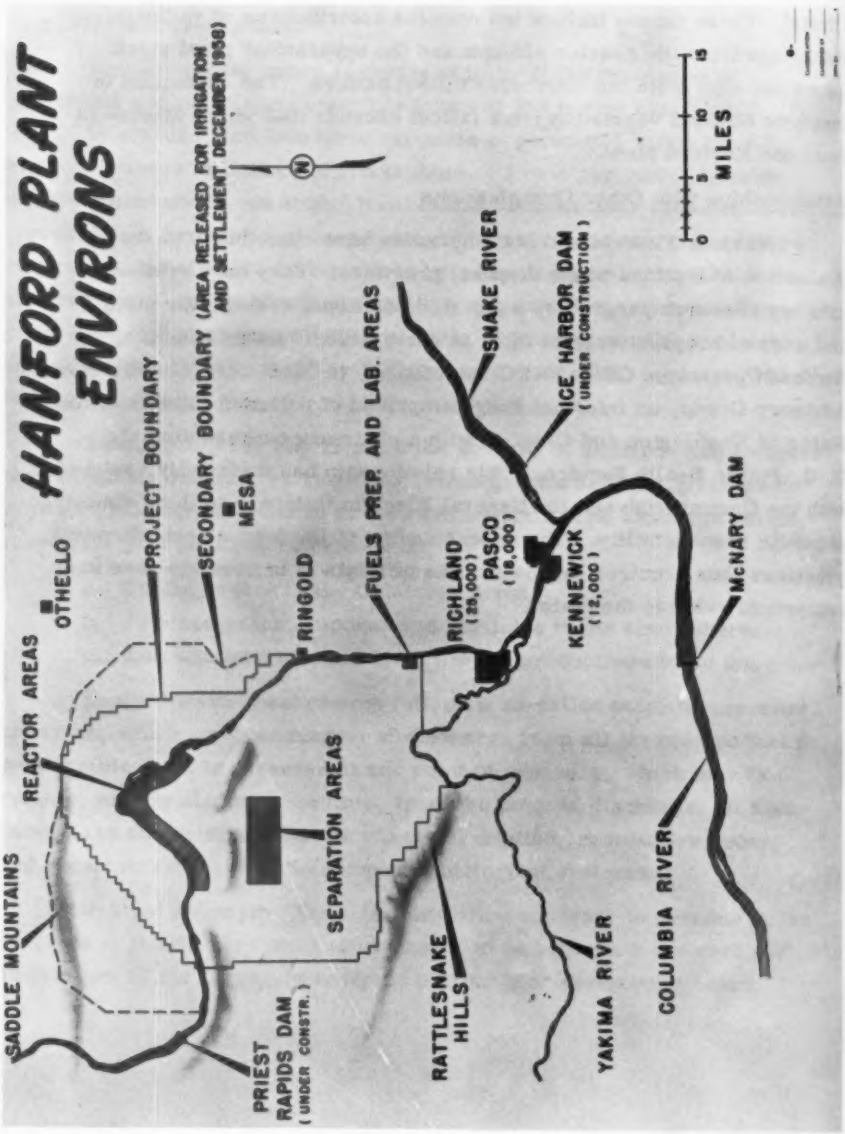

HANFORD PLANT
ENVIRONS
SADDLE MOUNTAINS
REACTOR AREAS
OTHELLO
PROJECT BOUNDARY
SECONDARY BOUNDARY (AREA RELEASED FOR IRRIGATION AND SETTLEMENT DECEMBER 1958)
MESA
RINGOLD
FUELS PREP AND LAB. AREAS
SEPARATION AREAS
RICHLAND (25,000)
PASCO (18,000)
KENNEWICK (12,000)
SNAKE RIVER
ICE HARBOR DAM (UNDER CONSTRUCTION)
McNARY DAM
PRIEST RAPIDS DAM (UNDER CONSTR.)
RATTLESNAKE HILLS
YAKIMA RIVER
COLUMBIA RIVER
N
0 5 10 15
MILES

EXHIBIT 2

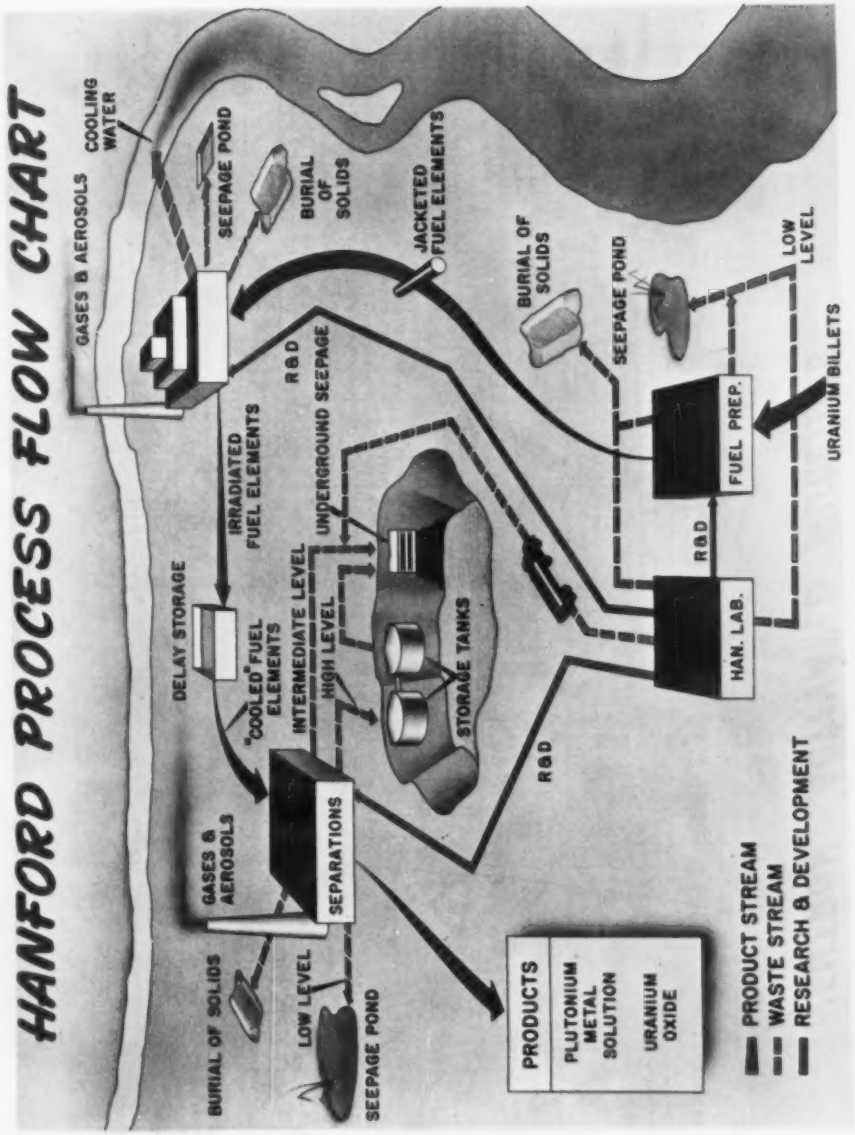

HANFORD PROCESS FLOW CHART
COOLING WATER
GASES & AEROSOLS
SEEPAGE POND
BURIAL OF SOLIDS
JACKETED FUEL ELEMENTS
BURIAL OF SOLIDS
SEEPAGE POND
LOW LEVEL
URANIUM BILLETS
R&D
IRRADIATED FUEL ELEMENTS
UNDERGROUND SEEPAGE
DELAY STORAGE
"COOLED" FUEL ELEMENTS
INTERMEDIATE LEVEL
HIGH LEVEL
STORAGE TANKS
GASES & AEROSOLS
SEPARATIONS
R&D
FUEL PREP.
HAN. LAB.
R&D
BURIAL OF SOLIDS
LOW LEVEL
SEEPAGE POND
PRODUCTS
PLUTONIUM METAL SOLUTION
URANIUM OXIDE
PRODUCT STREAM
WASTE STREAM
RESEARCH & DEVELOPMENT

EXHIBIT 3

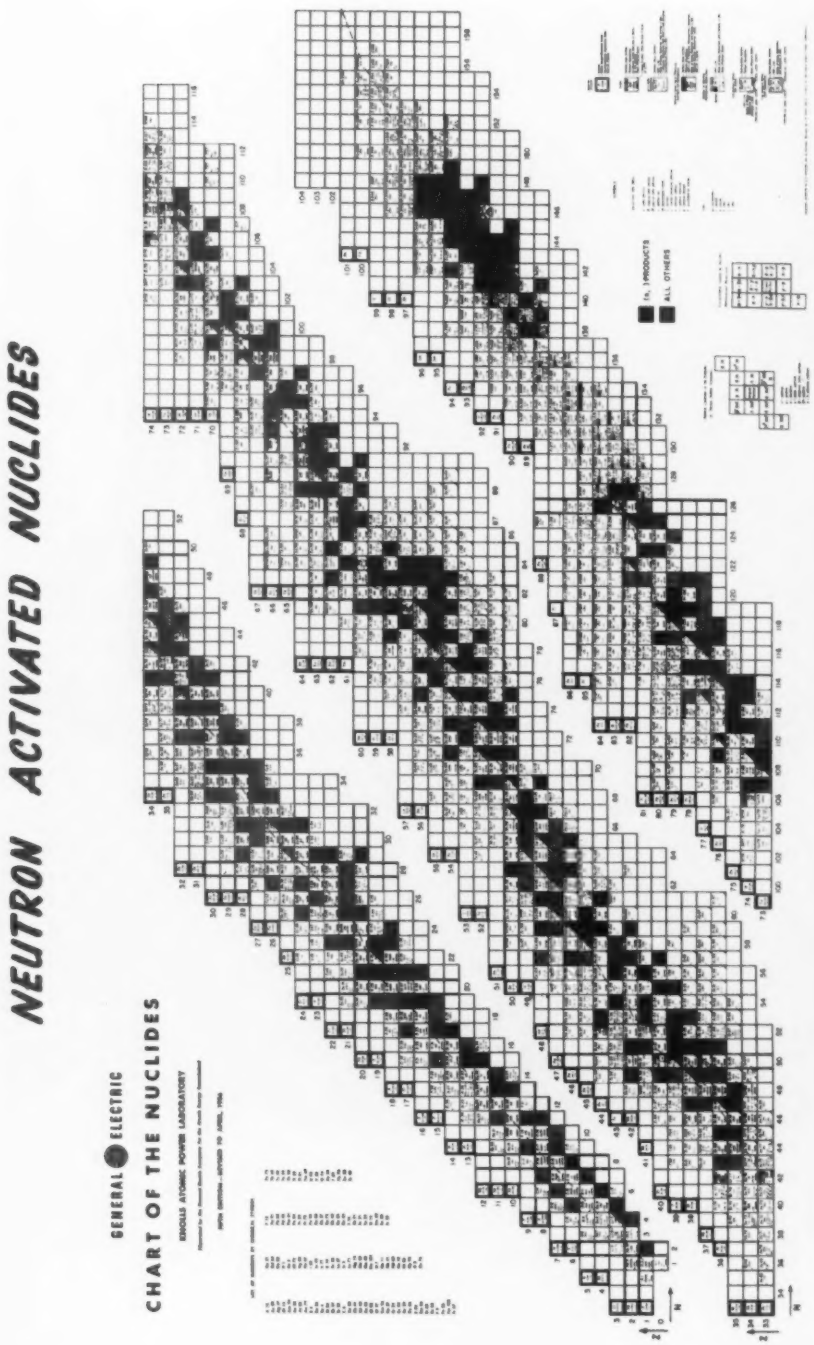

NEUTRON ACTIVATED NUCLIDES
GENERAL ELECTRIC
CHART OF THE NUCLIDES
KNOLLS ATOMIC POWER LABORATORY
FISSION PRODUCTS
ALL OTHERS

EXHIBIT 4

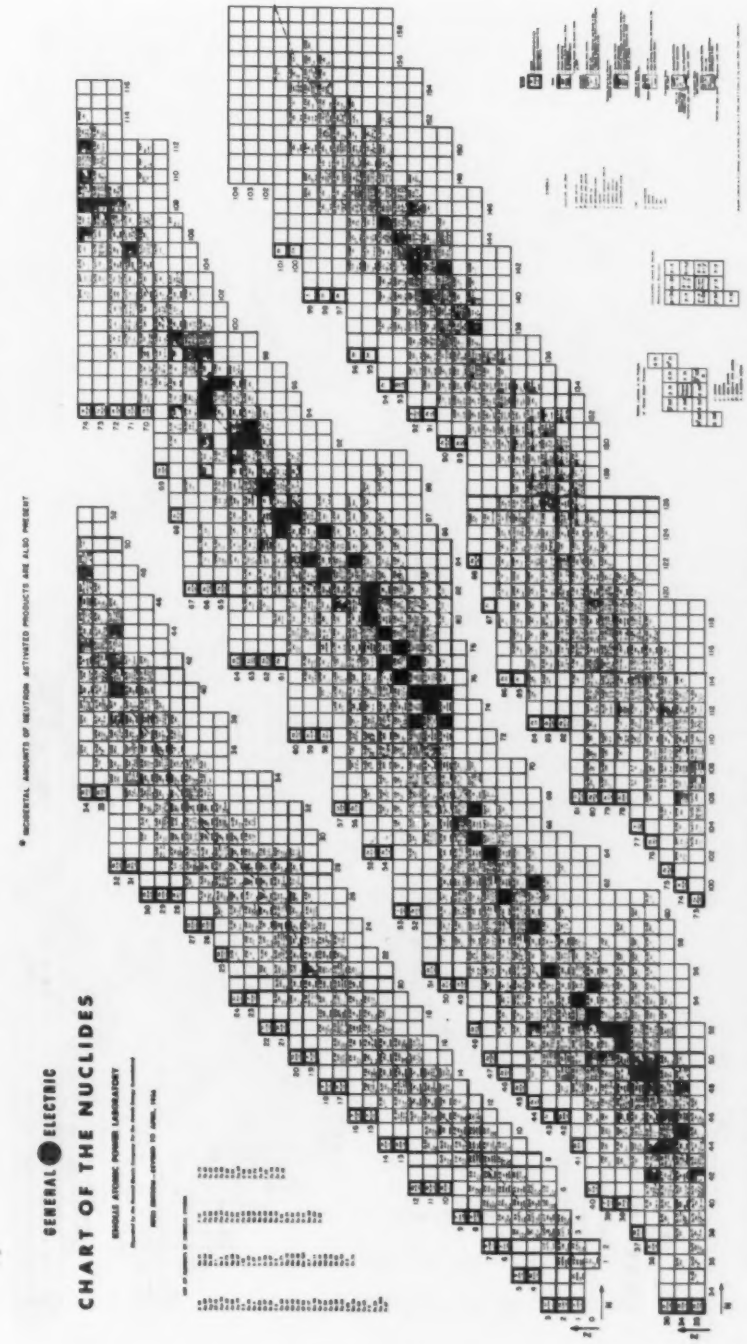

FISSION PRODUCTS OF IMPORTANCE IN SEPARATIONS AREAS
* INCIDENTAL AMOUNTS OF NEUTRON ACTIVATED PRODUCTS ARE ALSO PRESENT
GENERAL ELECTRIC
CHART OF THE NUCLIDES

EXHIBIT 5

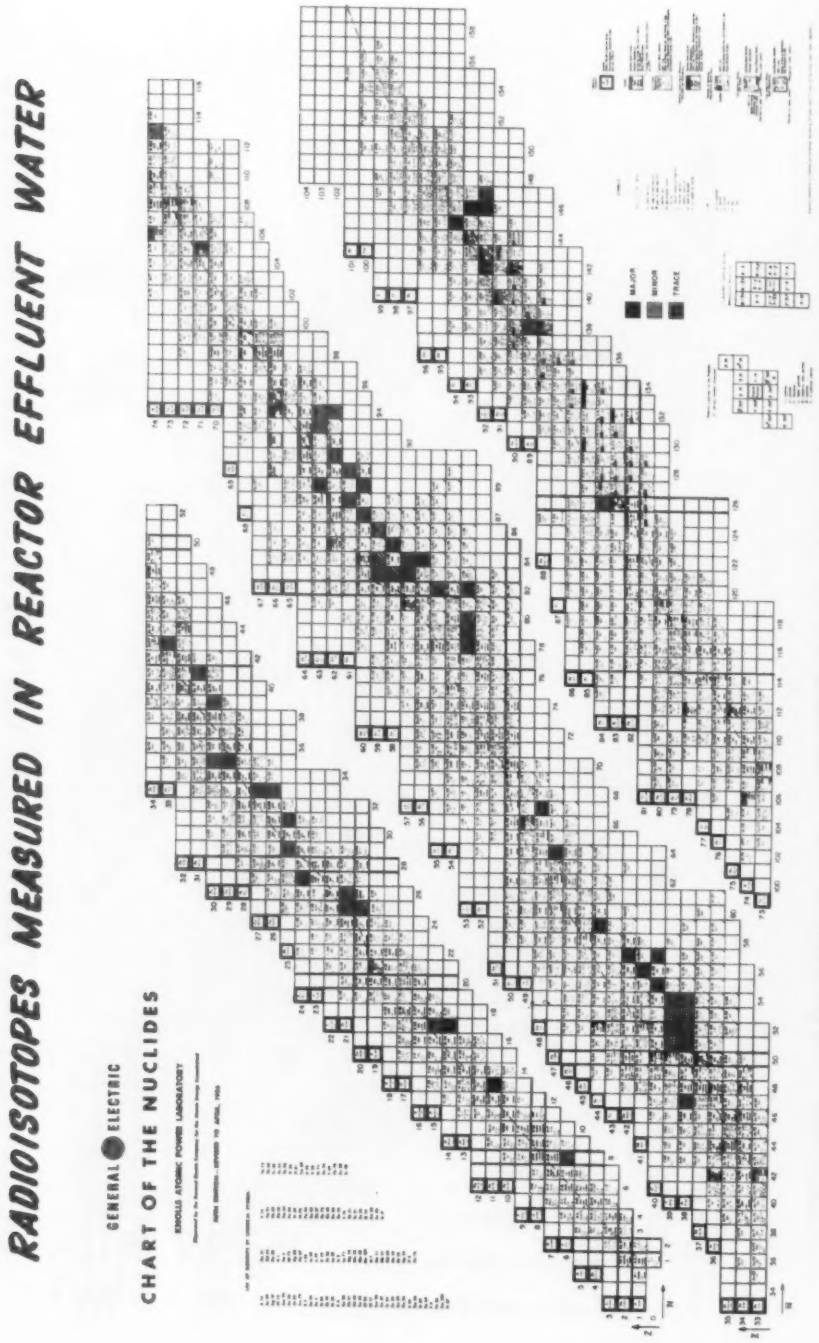

RADIOISOTOPES MEASURED IN REACTOR EFFLUENT WATER
GENERAL ELECTRIC
CHART OF THE NUCLIDES
KNOLLS ATOMIC POWER LABORATORY
MAJOR
MINOR
TRACE

EXHIBIT 6

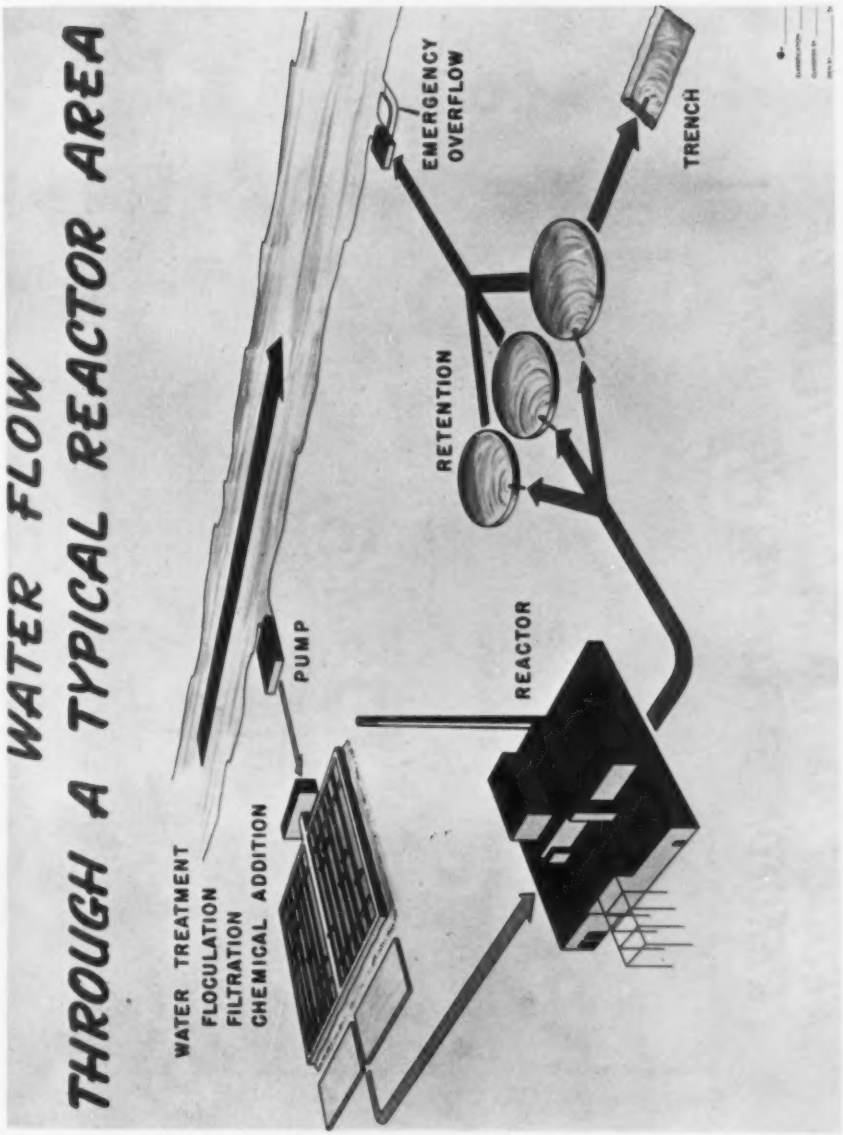

WATER FLOW
THROUGH A TYPICAL REACTOR AREA
WATER TREATMENT
FLOCULATION
FILTRATION
CHEMICAL ADDITION
PUMP
REACTOR
RETENTION
EMERGENCY
OVERFLOW
TRENCH

EXHIBIT 7

REDUCTION IN RADIOACTIVITY OF REACTOR EFFLUENT WITH TIME
* TRAVEL TIME TO THESE POINTS IS FOR A NOMINAL FLOW RATE OF 85,000 CU. FT. PER SECOND.
RETENTION IN BASINS
RINGOLD*
RICHLAND*
PASCO-KENNEWICK*
PER CENT OF ORIGINAL RADIOACTIVITY
100
80
60
40
20
0
0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30 32
HOURS AWAY FROM REACTOR

EXHIBIT 8

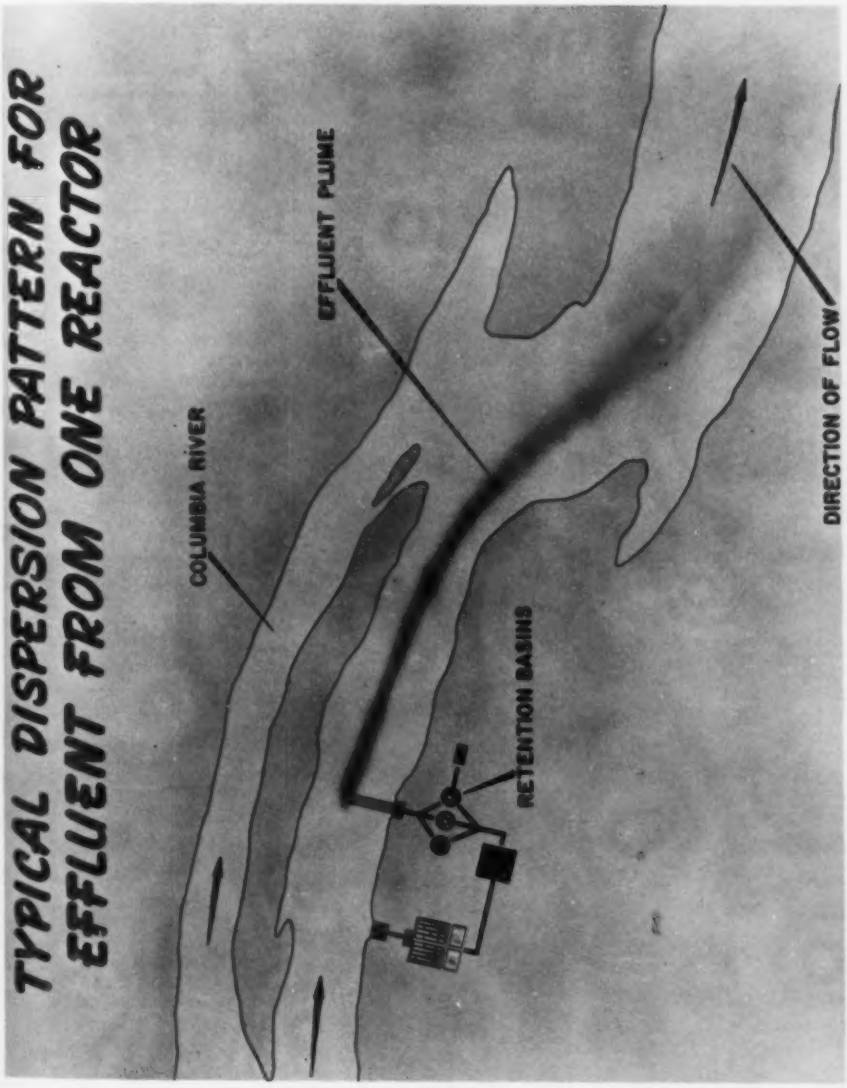

TYPICAL DISPERSION PATTERN FOR EFFLUENT FROM ONE REACTOR
COLUMBIA RIVER
EFFLUENT PLUME
RETENTION BASINS
DIRECTION OF FLOW
HOURS AWAY FROM REACTOR

EXHIBIT 9

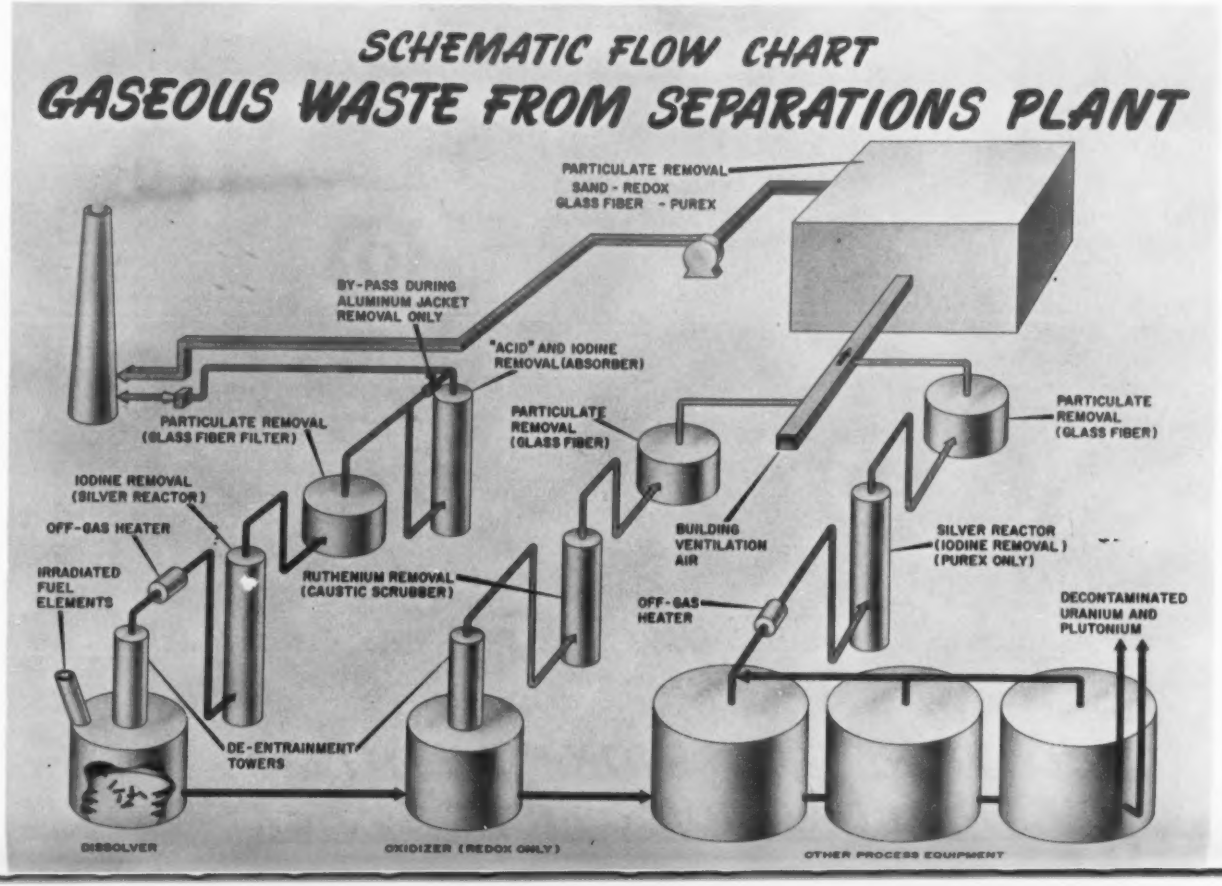

SCHEMATIC FLOW CHART
GASEOUS WASTE FROM SEPARATIONS PLANT
PARTICULATE REMOVAL
SAND - REDOX
GLASS FIBER - PUREX
BY-PASS DURING
ALUMINUM JACKET
REMOVAL ONLY
"ACID" AND IODINE
REMOVAL (ABSORBER)
PARTICULATE REMOVAL
(GLASS FIBER FILTER)
PARTICULATE
REMOVAL
(GLASS FIBER)
PARTICULATE
REMOVAL
(GLASS FIBER)
IODINE REMOVAL
(SILVER REACTOR)
OFF-GAS HEATER
IRRADIATED
FUEL
ELEMENTS
RUTHENIUM REMOVAL
(CAUSTIC SCRUBBER)
BUILDING
VENTILATION
AIR
SILVER REACTOR
(IODINE REMOVAL)
(PUREX ONLY)
OFF-GAS
HEATER
DECONTAMINATED
URANIUM AND
PLUTONIUM
DE-ENTRAINMENT
TOWERS
DISSOLVER
OXIDIZER (REDOX ONLY)
OTHER PROCESS EQUIPMENT

EXHIBIT 10

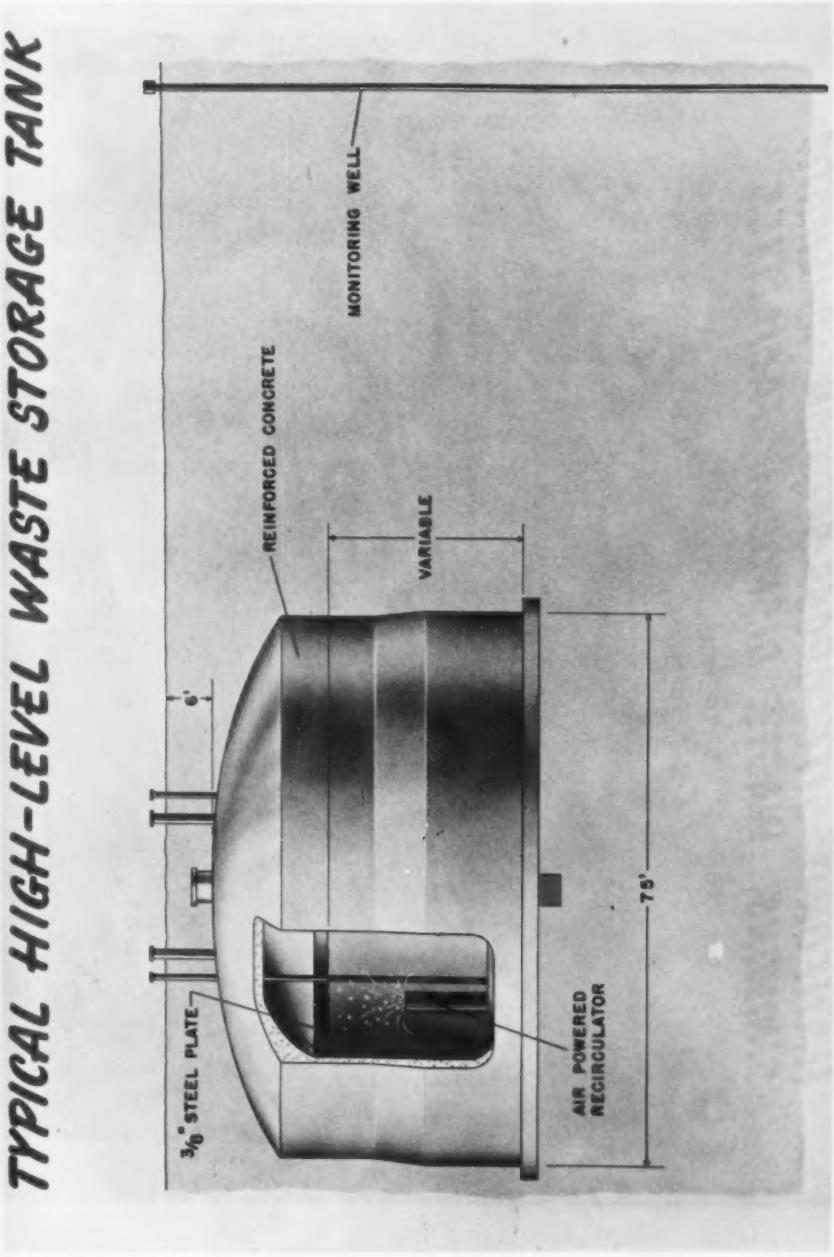
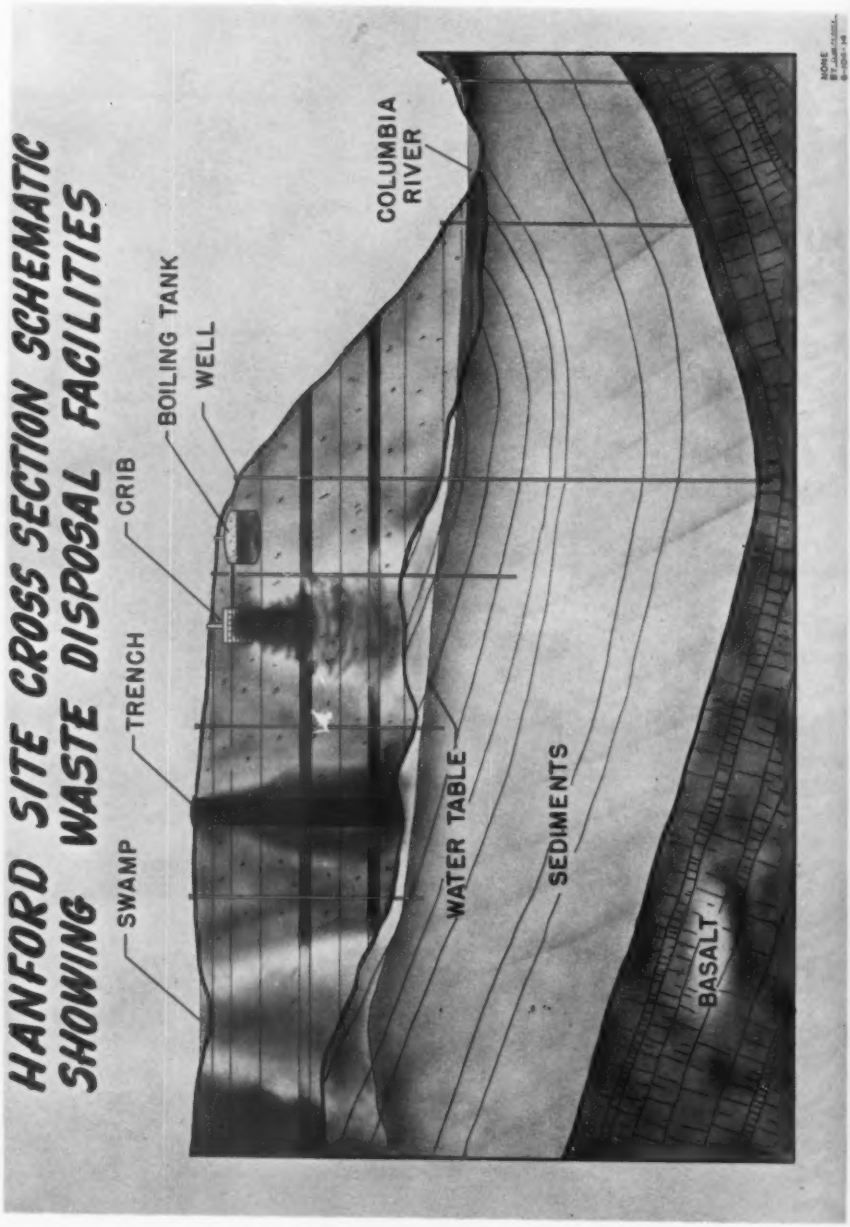

TYPICAL HIGH-LEVEL WASTE STORAGE TANK
MONITORING WELL
REINFORCED CONCRETE
VARIABLE
6'
3/8" STEEL PLATE
AIR POWERED RECIRCULATOR
75'

EXHIBIT 11


HANFORD SITE CROSS SECTION SCHEMATIC
SHOWING WASTE DISPOSAL FACILITIES
SWAMP
TRENCH
CRIB
BOILING TANK
WELL
COLUMBIA RIVER
WATER TABLE
SEDIMENTS
BASALT

ENVIRONS MEASUREMENTS (ANNUAL)

Total Samples	40,000	Biological Monitoring Samples	
Total Analyses	75,000	Birds	500
Field Measurements	40,000	Rabbits	700
Gaseous Waste Samples	10,000	Aquatic Organisms	1,200
Liquid Waste Samples	17,000	Total Test Wells since 1943	600
Vegetation and Ground			120,000 ft.
Samples	11,000		
Field Measurements	17,000		

EXHIBIT 12

SOURCES OF EXPOSURE

A. COLUMBIA RIVER

Drinking Water

- G. I. Tract
- Blood Stream
- Organ of Deposition

External Irradiation

- Swimming
- Boating
- Water Purification

Food Supply

- Aquatic Plants
- Fish
- Water Fowl
- Irrigated Crops
- Animals on Irrigated Land
  Milk - Eggs - Meat
- Food Preparation Plants

B.

EXHIBIT 13

SOURCES OF EXPOSURE

B. ATMOSPHERE

Breathing

- Lung Irradiation
- G. I. Tract
- Blood Stream
- Organ of Deposition

External Radiation from Cloud

Deposition

- Crop Contamination
- Uptake in Crops
- Uptake in Animals
  - Milk - Eggs - Meat
- Water Supplies
- External Radiation
- Direct Deposition on Man
  - Uniform Aerosol
  - Single "Hot" Particle

EXHIBIT 14

SOURCES OF EXPOSURE

C. GROUND DISPOSAL

Disposal Site

External Irradiation

Wind Pickup of Dried Materials

Uptake in Plants and Animals

Underground Water

Use of Wells

Deep-Rooted Plants

Travel to River

Re-Cycle into Chart on River System

EXHIBIT 15

Representative HOLIFIELD. I have for the record a description of the waste management operation at Hanford, prepared by the General Electric Co. staff.

(The material referred to follows:)

RADIOACTIVE WASTE MANAGEMENT OPERATIONS AT THE HANFORD WORKS

This paper was prepared by the staff of the Hanford Atomic Products Operation, General Electric Company, and filed for the record of these hearings upon the request of the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy.

By the Staff of the Hanford Atomic Products Operation

General Electric Company

PURPOSE AND SCOPE OF THIS DOCUMENT

In connection with Public Hearings on Industrial Radioactive Waste Disposal before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, oral testimony has been prepared to describe the management of such wastes at the Hanford Atomic Products Operation. A principal purpose of this document is to provide supporting information in more technical detail for the oral statement.

The format of this document is arranged for convenient reference to the several facets of radioactive waste management at Hanford, viz:

Part 1: A general scoping of the waste disposal management by H. M. Parker, Manager, Hanford Laboratories with identification of four major areas of waste disposal operations, which are then detailed as follows:

Part 2: Release of reactor cooling water to the Columbia River by R. F. Foster, Manager, Aquatic Biology, Hanford Laboratories Operation.

Part 3: Release of gases, vapors, and particles to the atmosphere, by R. E. Tomlinson, Manager, Advance Process Development, Chemical Processing Department.

Part 4: Storage of high-level fission product wastes also by R. E. Tomlinson

Part 5: Disposal of intermediate and low-level wastes to the ground by D. W. Pearce, Manager, Chemical Effluents Technology, Hanford Laboratories Operation.

Two notable characteristics of the Hanford Atomic Products Operation program on waste disposal have been:

1. Close integration of operations with research and development.
2. Wide diversity of professional disciplines in that research and development.

The authors of Parts 1-5 have drawn freely from the work of associates in many other disciplines. This is partially recognized in the bibliographies and is here acknowledged. In addition, J. W. Healy, P. C. Jerman, A. R. Keene, and H. V. Clukey made substantial contributions to the marshalling of the information presented. To these and others too numerous to mention, the present authors wish to make suitable acknowledgement.

The scope includes also a Glossary of Terms, Part 6, relating to radioactive waste disposal as practised at this location. Many of these terms are identical with national usage and they are included for the convenience of those readers who have not had occasion to keep in close touch with nomenclature in this rapidly developing field.

Part 7 is a bibliography of Hanford publications in this field with a brief abstract of those whose relevance is not obvious by title. It should be noted that many of the referenced documents are obsolete and superseded, and are not recommended for use out of the present context. For example, some are included to show progressive stages of sophistication with respect to validating particular permissible limits. The bibliography includes classified documents in the category which theAtomic Energy Commission has released under its Civilian Application Program. Certain other documents necessarily of higher classification may be made available to the Joint Committee upon request.

Because of the aforementioned integration between operations and research and development, the Hanford position is not adequately presented without reference to the local research and development.

Hanford Atomic Products Operation personnel are submitting relevant material on certain topics in Section IV of the Hearings, as follows:

Surface waterways	R. F. Foster
Ground disposal	D. W. Pearce
Meteorologic studies	G. R. Hilst
Removal of specific isotopes	R. E. Tomlinson

These documents should be considered together with this contribution.

PART 1

HANFORD RADIOACTIVE WASTE MANAGEMENT

H. M. Parker

Introduction

The Hanford plant is located in the Southeastern portion of the State of Washington. It is one of the nation's major plutonium-producing complexes, operated, since 1946, by the General Electric Company under contract to the United States Atomic Energy Commission.

The general location of the area is shown in Figure 1. The site was acquired in 1943 by the U. S. Government. It consists of a primary area of 429 square miles bounded on the north and east by the Columbia River and on the south and west by a perimeter fence, with a supplementary primary area of 138 square miles. Some 164 square miles on the north side of the river which had been held as a secondary area, was released for irrigation and settlement in December of 1958. The site is approximately a rectangle 30 miles north to south and 25 miles east to west. On the north, west, and south sides are mountains of 2,000 to 4,000 feet elevation. Rainfall averages about 7-1/2 inches per year, the natural ground cover is sagebrush and grass, and the surrounding farmlands are either irrigated for produce and pasture, or used for dry-land farming at the higher elevations.

Eight plutonium-producing reactors are located along the river in the northern part of the primary area at an elevation of about 400 feet above sea level. The various chemical separations plants are on an interior plateau of 600 to 700 feet elevation, about ten miles from the river. In the southeast corner of the project are the majority of the research laboratories and the fuel preparation facilities.

(For figure 1, referred to above, see exhibit 1, p. 184.)

A few miles south of the reserved area is the city of Richland, which recently became incorporated. Property within the city is currently being disposed of by the Atomic Energy Commission. The administration area for the General Electric Company and the Atomic Energy Commission is located in Richland. Five miles north of Richland is Camp Hanford, base of the U. S. Army personnel guarding the project. About 80,000 people live in the region around the plant perimeter. The sanitary water for the neighboring cities of Pasco and Kennewick is derived from the Columbia River. Almost without exception, wells supply the remainder of the populace.

Hanford Operations

Initial construction and operation of the Hanford plants was performed by the DuPont Company in the period 1943 to 1946. Since September, 1946, the General Electric Company has been the prime contractor for Hanford operations. Addition of reactors and replacement of obsolete separations plants have followed the original location patterns as the wartime construction site evolved to a stable position in the atomic energy program. Major research and development facilities were also added to provide bases for technological improvements in the process and in the radiological fields.

The following description of operations, wastes sources and environmental concentrations is intended to provide a broad introduction to the waste management at the site. Four principal areas of activity are identified. These are described in more detail in Parts 2-5.

Fuel Preparation

The plutonium production process begins at Hanford with the canning of uranium fuel elements in pre-formed aluminum capsules. Because the fuel preparation is a relatively low risk operation, similar to other common heavy metal industries, the plant is conveniently located near the residential area.

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The problems of radioactive waste disposal from fuel preparation are insignificant compared to those of later processes. The purified, unirradiated natural uranium metal received at Hanford is radioactive and has to be handled with respect. However, upon irradiation in the nuclear reactors its radioactivity is increased by a factor of several million. This ratio of several million between the radioactivity before and after irradiation indicates the relative magnitude of waste disposal problems from the fuel preparation activities as compared with later processes. Cooling water, pickling rinses, and dilute caustic liquid wastes from the canning process, together with depleted uranium waste solutions from pilot plant studies, are discharged to either of two surface ponds near the Columbia riverbank. Uranium in the pond water averages about 5×10^{-7} microcuries per cubic centimeter in concentration which is about 25 per cent of the maximum permissible concentration for occupational exposure. The pond water is not consumed by humans. First it percolates down through the soil to the water table, merges with the groundwater, and finally is greatly diluted as it drains into the river. Uranium is found naturally in the Columbia River water at a concentration of about one microgram per liter ($\sim 7 \times 10^{-10}$ /uc/cc). (At an annual average river flow of about one million gallons per second, this means that about 130 tons of uranium go down the Columbia River each year.) Only for a distance of about one mile downstream from the ponds can the additional uranium from the fuel preparation activities be detected above the natural uranium found in the river water.

The soil and groundwater throughout this region also contain appreciable natural concentrations of uranium. Water samples from wells on the Pasco side of the river, at Benton City to the west, and from the Yakima River show concentrations 5 to 15 times that in the Columbia River. These concentrations are in no way connected with past operations of the Hanford plant.

In summary, wastes from the fuel preparation plant do not add distinguishable concentrations of radioactive material to those naturally present in the environment.

Reactor Operations

The eight nuclear reactors for producing plutonium from uranium are located along the Columbia River in the northern portion of the reservation. This location permits free access to the copious quantities of cooling water required to dissipate the heat generated in the process. It also provides maximum isolation of the facilities from the more densely populated communities.

In the irradiation process, the canned fuel elements are subjected to intense neutron bombardment. Part of the uranium is transmuted into the desired product, plutonium. Another part splits by the fission process into numerous elements collectively known as the fission products. These products are intensely radioactive. At this stage they are physically retained in the fuel jackets. Both their heat and the energy of the nuclear fission itself are carried away by cooling water.

The cooling water pumped from the river is purified by coagulation, sedimentation, and filtration before it is used in the reactor. During its passage through the reactor tubes, traces of minerals in it are made radioactive. The effluent is retained for a period of one to three hours in basins or tanks to allow the major portion of the induced radioactivity to decay. Recent studies have indicated that the elimination of this retention period would have little, if any, deleterious effect on the river. After this retention period the water is discharged directly to the bottom of the river channel through large pipes anchored to the river bed. The location of the reactors well away from residential areas allows

decay of most of the residual radioactivity of the effluent before the water flows downstream to zones of public use. This remoteness also provides a dispersion zone for the minor amounts of airborne radioactive contaminants.

The radioactive wastes from the reactor areas, in order of their environmental importance, include:

(1) Discharge of the normal cooling water (effluent) to the Columbia River,

(2) Discharge of small volumes of abnormally contaminated effluent water to trenches along the riverbank,

(3) Discharge of gases and particulates to the atmosphere,

(4) Burial of solid wastes in the ground.

The radioactive materials in the effluent are primarily activation products produced in the reactors by the capture of a neutron in a normal element. Trace quantities of fission products also present result from neutron bombardment of the uranium naturally present in the river water, and from occasional fuel element failures. The activation products are present in quantities beyond those expected from neutron capture by impurities in the water during passage through the reactor. This is because holdup of the mineral elements in a film on the inside of the reactor tubes permits a longer exposure to the neutron flux. In addition, some isotopes, such as sodium-24, are contributed in part by the slow corrosion of the reactor tubes and the aluminum jackets of the fuel elements. Table 1 indicates the isotopes which have been detected in the effluent water.

Table 1

Reactor Effluent Isotopes in Order of Decreasing Concentration, Four Hours after Irradiation

Major (90%)	Minor (8%)	Trace (2%)		
Mn^{56}*	Zn^{69}	Eu^{152}	I^{131}*	Pr^{145}
Cu^{64}*	Ga^{72}	Sm^{153}	Ce^{141}	Pm^{151}
Na^{24}*	Sr^{92}	W^{187}	Pr^{142}	Co^{60}
Cr^{51}*	U^{239}	La^{141}	C^{14}	Pr^{143}
Np^{239}*	I^{133}*	Nd^{149}	Nd^{147}	Ru^{103}
As^{76}*	Y^{92}*	La^{140}	Ca^{45}*	Sc^{47}
Si^{31}*	Nb^{97}*	I^{132}*	Ag^{111}	Sr^{90}*
	Sr^{91}*	Eu^{157}	Y^{91}	Cs^{137}
	Zn^{65}*	Ba^{140}*	Fe^{59}*	Sr^{85}
	P^{32}*	Mo^{99}	Sr^{89}*	U^{238}*
	Y^{90}*	Sm^{156}	Mn^{54}	Pu^{239}*
	I^{135}*	Sc^{46}*	Zr^{95}	Ac^{227}
	Y^{93}	Cd^{115}	Pm^{149}	Po^{210}*
		Ce^{143}	Eu^{156}	

*Routine measurements are made on these isotopes.

The relative rankings of these isotopes in terms of radiation exposure and the reasons why that ranking is not necessarily the same as the order of abundance will be discussed in Part 2.

The higher power levels and changes in water purification which have evolved throughout Hanford's history have caused slight changes in the types of radioactive materials in the discharged cooling water. In quantity, there has been a substantial increase since 1944. Careful measurement of the characteristics of the released material has provided assurance that these increases could be made without reaching the relevant maximum permissible limits.

Small volumes of water which contain higher concentrations of radioactive materials than the normal effluent arise from certain operations. Although the total amount of radioisotopes resulting from these operations is small in comparison with the quantities in the effluent, these wastes are sometimes disposed of to trenches along the banks of the river. Here the water percolates into the river through the soil with consequent filtering of particles and ion exchange of the soluble matter.

At the present time there are some 50 separate locations in the reactor areas where such disposal has been carried out; about 25 of these locations are still in use. It is not necessary to select locations that assure complete removal of all radioactive material before these wastes reach the river.

Gaseous and particulate wastes are released through high stacks to the atmosphere. The radioactive materials present in this air arise from dusts picked up by the ventilation air from normally unoccupied portions of the building and from leakage of the gas used in the reactor chassis to exclude air and moisture. Minor quantities of radioactive materials are released from these stacks in normal operation. They produce no distinguishable contamination levels in the air or ground outside of the restricted area. Minor ground contamination, restricted to the vicinity of a reactor, has resulted from the few abnormal operating incidents to date. Equipment to filter and scrub the stack gases before release is now in the design stage.

Solid wastes, such as contaminated paper, boards, worn-out tools, construction items, and aluminum spacers used to center the fuel charge are buried in shallow trenches in the reactor areas. Such buried wastes are isolated from the environment since care is taken that the bottoms of the trenches are above the water table, even during river flood stages, and the rainfall is low enough so that percolation through the deposits is practically non-existent.

In summary, the large volume of cooling water which contains relatively low-level concentrations of activation products caused by irradiation of the water in the reactors, is the principal waste disposal consideration in reactor operations. The routine releases of gases and particulates from the ventilation systems are and will continue to be minor except in the case of a substantial reactor accident. Such an accident, of course, has not been experienced at Hanford.

Chemical Separation Operations

The fuel elements are irradiated in the reactors until the desired plutonium content is obtained. The elements are then discharged and stored underwater for a few months to allow the radioactivity to decrease to the level which can be handled in the shielded chemical separation facilities. This aging is commonly referred to as the cooling period. The aluminum cans are removed and the fuel elements dissolved in nitric acid. At this stage various radioactive gases such as radio-xenon, vapors such as radio-iodine, and particles are released. Some of these air-borne constituents are removed by specific processes and the treated air is released through high stacks. The principal residual contaminant is I^{131}. The appropriate release quantities have been the subject of intensive study at Hanford. Part 3 is concerned with the management of these wastes discharged to the atmosphere.

The uranium and plutonium are separated from the fission products and purified by solvent extraction processes. The depleted uranium is converted to a suitable powder for off-site shipment. The plutonium is further purified and prepared for the end applications. The so-called high level wastes contain most of the fission products and these are sent to million-gallon underground tanks for storage.

At Hanford, the terms high, intermediate, and low level wastes have meanings which are relevant to the local processes and environment. At other sites these terms obviously could have different connotations. Local terminology is described more fully in Part 6.

High level liquid wastes at Hanford contain radioactive materials from the principal chemical separations processes. They are all retained indefinitely in underground storage tanks. The concentration of radioactive isotopes is generally greater than 100 /uc/cc, and for the most part is

orders of magnitude greater than this. These wastes result from the first decontamination of the uranium and plutonium and contain the bulk of the fission products. The volume of high level wastes stored in underground tanks is currently about 52,000,000 gallons. Placement of the tanks well below ground serves the dual purpose of shielding personnel from their radiation, and protecting the tanks themselves. It is generally considered that the method is storage rather than disposal. No environmental hazard exists as long as the tanks maintain their integrity. The position of the tanks on the interior plateau with some 200 feet of dry sediments below means that a modest leak would not create a significant hazard. Moreover, experience in connection with removal of certain constituents of old wastes has shown that the wastes can be readily transferred to new tanks. Furthermore, a considerable part of the radioactivity becomes associated with a heavy sludge that should have little inclination to travel underground.

The sensible management of these wastes thus amounts to maintaining the underground tank storage until such time as

(1) A cheap proven method of ultimate disposal is developed,

or (2) A reliable method of assuredly immobilizing the wastes in place is developed,

or (3) A profitable use as a byproduct is developed.

In recent years programs of self-concentration and scavenging of wastes followed by ground disposal of the purified liquid have recovered storage tank space so that additional tank construction has been postponed. This sets a reasonable limit to the ultimate volume of stored wastes. Part 4 gives an account of the control of these high level wastes.

Intermediate level liquid wastes are those which require additional decontamination before release to the environs but do not require long storage times in tanks. Under Hanford conditions the concentrations in

this category range from about 10^{-5} to 100 /uc/cc. Separations process wastes in this category include secondary decontamination streams, process condensate from high-level waste concentration, and scavenged high-level waste supernates. Organic or inorganic chemical content incompatible with adequate interaction with soil (e.g., ion exchange) may require tank retention of some wastes in this category.

Due to the favorable geological and hydrological conditions at the separations area it is possible to retain the vast majority of the radioactive materials in the approximately 200-foot layer of sediments between the ground surface and the local water table. Here the ion exchange capacity of the small clay fraction of the soil causes the radioisotopes to be adsorbed and retained in the vicinity of the disposal site. Since the rainfall is low in this region, only minimal leaching or transfer by other mechanisms is encountered and the radioactive materials are essentially "stored" in the sediments above the water table. The water percolating to the water table is purified by this mechanism. Additionally, decay time and ion exchange capacity would be available during the relatively slow movement of the groundwater to the Columbia River where copious dilution would be available. Each major type of waste is evaluated in the laboratory for its behaviour in the soil before disposal operations are started. After disposal is under way, a program of monitoring is maintained to note the travel of isotopes through the ground and any transfer to the groundwater. In this program, any process waste stream which would not be properly retained in the soil is treated as a high-level waste and stored in tanks.

Intermediate level wastes enter the ground by seepage from structures known as cribs or caverns and trenches. The cribs or caverns are buried structures from which the liquids can percolate into the ground. Open trenches are used for single disposal operations with the trench refilled immediately to prevent dissemination of surface contamination by winds.

Over three billion gallons of intermediate level wastes have been admitted to the ground.

Separations process cooling water and utility steam condensates are discharged to surface ponds or swamps. Since these waters may occasionally be slightly contaminated by process vessel leaks, they are considered in the category of other low-level wastes, with concentrations less than about 10^{-5} /uc/cc. Nearly thirty-five billion gallons of water have been handled in this manner.

The large volume released to surface ponds has built an artificial mound of water on top of the original water table under each of the two separations areas. The western mound, in fine sand and clay, stands about 70 feet high; the eastern mound, in more porous sand and gravel is spread out and stands about 20 feet above the original water table. These mounds substantially modify the natural hydraulic gradients and so may alter the paths of any radioactive material which reaches the groundwater.

At the present time, the mounds are partially advantageous because they form, as it were, a water dam opposing the movement of possibly contaminated liquid toward the river. However, it would not be prudent to predicate intermediate level waste disposal practice on this effect. If one did, the mounds would have to be maintained long after the plants might be shut down, or new processes not generating liquid wastes developed. Part 5 discusses the interaction of research and development and of operating practice in the effective control of both intermediate and low level wastes.

The need for retaining separations plant waste within the project, while reactor effluent cooling water may be discharged to the public waterways is illustrated to a first approximation by the relative hazards if each waste were to be drunk by humans. The radioactive material in one gallon of typical intermediate level separations plant waste is equivalent in radiological hazard to the radioactive material in 20,000 gallons of reactor

effluent discharged to the Columbia River. The radioactivity in the reactor effluent decays so rapidly that 24 hours later this same one gallon of separations plant water would be equivalent to one million gallons of reactor effluent.

The dissolving of the fuel elements in nitric acid was briefly mentioned as the principal source of releases of material to the atmosphere. More of the same materials are released in the later processing steps. These elements are removed by chemical scrubbing and filtration and the air released to the stacks. The ventilation air of the process buildings is also vented to high stacks because it may, from time to time, contain minor releases from process vessels. It is passed first through large filter beds which remove virtually all of the entrained radioactive particles. Numerous sampling stations in this train of equipment permit a continuous recording of the releases. In working practice, it is the I^{131} which predominates as the material requiring control.

In summary, the major effort on radioactive waste treatment at Hanford is expended on the liquid and gaseous wastes from the chemical separations plants. Complete isolation from the surrounding environs, by retention in tanks or the ground, is the method of handling the radioactive material in liquid wastes from these separations areas. Treatment of the air to remove most of the contaminants, followed by controlled releases of the remainder so that concentrations are within maximum permissible environmental limitations, is the method of handling radioactive material in gaseous wastes. This limitation of environmental contamination is accomplished by atmospheric dilution of the noble gases released through 200-foot stacks, filtration of particulates, conversion of most of the radioactive vapors to liquids which are retained in tanks or the ground, and dispersion of residual vapors to the atmosphere through the stacks.

Research and Development Operations

Research and Development is conducted at the Hanford Atomic Products Operation in virtually all fields of the atomic energy program directly related to the Hanford production complex, and in a few selected additional fields. The major laboratory facilities are located in the Laboratory Area, a few miles from Richland. However, there are important experimental facilities located in both the reactor and separations areas, since in some cases it is impossible or impractical otherwise to simulate production conditions. For example, the rapid decay of reactor effluent radioactivity requires that the pertinent biology research facilities be located in a reactor area.

Relatively small amounts of process materials and other radioactive materials are handled in the laboratory facilities for research in chemistry, physics, metallurgy, ceramics, etc. The radioactive wastes from the laboratory facilities are generally similar to process wastes, but of smaller volume and usually of lower activity. In the reactor and separations plant areas, these laboratory wastes are discharged along with process wastes of similar composition. In the Laboratory Area, where the majority of the research facilities are located, liquid wastes such as cooling water, sink drainage, etc., which are expected to be uncontaminated, or those contaminated only with unirradiated uranium, are disposed to ground locally. They are held up in retention basins for sampling and analyses before disposal to trenches on the riverbank. Intermediate level wastes are collected and hauled by tank trailer periodically to the separations plateau for underground disposal with compatible wastes. High level wastes are mixed in concrete castings and buried a few miles from the river.

Exhaust air from laboratories is filtered before it is vented to the atmosphere from short stacks on laboratory roof tops. The exhaust air from these laboratory facilities discharges negligible amounts of radioactive materials to the environs.

In summary, the discharge of radioactive wastes from research and development activities, while requiring due care, is insignificant in the over-all Hanford waste disposal picture.

Bases of Radiation Protection at Hanford

Working radiation protection standards and practices at Hanford are established, and modified as appropriate, by a separate Company component known successively as Health Instruments Division, Radiological Sciences Department, and currently, Radiation Protection Operation of the Hanford Laboratories. Actual performance against these standards is also monitored by this component.

Responsibility for the conduct of operations with radioactive material to conform with such standards rests wholly with the respective managers of the production operations. Operating procedure manuals of such components incorporate the standards for application to the particular operations of the component. Radiation monitoring functions in production and laboratory facilities are unified through the standards, central instrument service and calibration, and centralized personnel exposure metering and record system. This duality assures the maintenance of a high quality of protection through a conventional system of checks and balances.

The local standards are not less conservative than those prescribed by the Atomic Energy Commission and are broadly responsive to those promulgated by other qualified sources such as:

International Commission on Radiological Protection

National Committee on Radiation Protection and Measurements.

Because of the historical trends toward the decreasing of limits applicable to radiation protection, and because of some uncertainties in the interactions of man, the animal kingdom, and plant life with radiation,

Hanford early and voluntarily assumed a conservative position on radiation protection. That the policy was indeed sound and not unduly conservative is shown after 14 years of operation when national and international limits were recently reduced to levels broadly similar to those in use at this site for many years.

A conservative posture will continue to be appropriate as long as it is essentially impossible to neutralize or subtract from man or his environment the radiation exposure already received.

Those sections of the manual of Radiation Protection Standards and related material which refer to radioactive waste disposal practices and responsibilities are introduced into the record as an appendix to this document. Detailed comparison with any particular national or international body of recommendations or regulations may thus be made.

Hanford has often held a position of national leadership in radiation protection as exemplified in the release of iodine-131 to the atmosphere. Control, if governed only by the inhalation risk to man, would lead to intolerable radiation damage to range animals due to radio-iodine deposition on vegetation and subsequent consumption by animals. Many of these potentialities were foreseen, and suitable research was started to establish proper standards in this broader concept of relevant hazard definition. The Hanford and national recommended limits are compared below:

Hanford I^{131} in air	about 1×10^{-13} /uc/cc
(based on deposition on vegetation of 1×10^{-5} /uc/gm)	
National I^{131} in air, based on humans outside of a controlled area	6×10^{-10} /uc/cc

Because the research and development at Hanford have been closely integrated with operating practise, through modification either of the formal internal standards or of operating procedures, it is now difficult

to make a satisfactory separation. Appreciation of the rationale of the Hanford position will require a joint consideration of the material presented here and that presented elsewhere under the topic of Research and Development.

Radioactive Waste Disposal Policy

Radioactive waste disposal policy at Hanford has developed hand-in-hand with changes in national recommendations on radiation exposure limits, with basic and applied research on biological and environmental effects, with advances in instrumentation and analytical methods, and with expansion and modification of the Hanford plants.

Basic waste disposal policy involves two broad concepts, viz:

1. Concentrate and contain
2. Dilute and disperse

Both are practised at Hanford. Where feasible, concentration and containment is to be preferred. However, auxiliary processes tend to generate such large volumes of low level wastes that containment is impractical. At the present time, release of the lower order liquid wastes to ground is necessary, as is the release of air, from which one has attempted to remove for containment as much of the radioactive content as is feasible. In going to dilution and dispersal of such wastes it is mandatory to control the amounts so that the safety of not only the human population, but also the plant and animal resources is assured.

Hanford uses a "point-of-exposure" philosophy which involves measuring and controlling the discharge of radioactive material to the extent necessary to insure that no undue radiological exposure is created at the significant point of exposure (which may be remote in both time and distance from the initial discharge). It is tantamount to recognizing the critical system in the complex chain of events following release.

This approach is allowed and limited by the use of available knowledge of geological, hydrological, meteorological and other environmental conditions. It recognizes and incorporates the effects of dilution, radioactive decay, and reconcentration of radioisotopes in aquatic organisms and food chains. With application of such knowledge, the "point-of-exposure" philosophy is considered more accurate and economical than the inflexible "point-of-discharge" control concept. Ultimately, of course, it is simpler to operate a plant with a set of engineering standards defining permissible releases. Ultimately, too, the various ramifications of dispersion and reconcentration may be sufficiently generalized to permit this approach. Its premature application could result in either severe handicaps to the atomic energy program, or alternatively, in potential radiological hazard following an unforeseen concentration mechanism.

Radioactive waste disposal operations within the project are limited by occupational exposure of the project personnel involved. The nationally recommended occupational exposure limits are used as the basis for control of Hanford personnel exposures.

Exposure of people outside of the project boundary is limited to the national recommendations for people in the vicinity of a controlled area. These limits are generally expressed as one-tenth of the corresponding maximum permissible limits for occupational exposure.* In addition, Hanford has

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* The language of some proposed recommendations calls for even lower limits when applied to the whole population. Presumably this arises from the need, not related to a given individual's welfare, to preserve a low rate of radiation induced genetic mutation. It requires to be examined separately for each specific isotope. Most of the radioisotopes make a very small contribution to gonadal dose compared with other significant organs.

self-imposed limits, such as those for iodine-131 on vegetation and phosphorus-32 in fish (see Part 2) which are substantially more restrictive. Because of the lag between release and exposure measurements, uncertainties in representative sampling and measurement accuracy, and shortcomings in the recommended biological limits, further restrictions are imposed for control purposes in certain cases. For example, operating control for radioisotopes in the Columbia River has been set at one-twentieth of the occupational exposure limit. Ground disposal at any one disposal site in the separations areas is to be stopped when the concentration of any radioisotope (of half-life greater than three years) in groundwater samples from that disposal site exceeds one-tenth of the maximum permissible concentration (occupational) in drinking water. This has proven to be economically possible and contains safety factors since no allowance is given for ion exchange in the groundwater sediments nor for decay during slow movement for several miles to the river, nor for the ultimate dilution in the river.

Finally, waste disposal is limited by the best estimates of the total exposure to all sources of environmental radiation, rather than by control of each waste stream to its maximum permissible exposure limit or each radioisotope to its maximum permissible concentration.

Transfer Mechanisms of Released Wastes

When radioactive material is released to the environment, there is an obligation to determine by broad surveys the various pathways of potential radiation exposure of man and animal or plant resources. In a given case, most of the available pathways lead to trivial end-results. Some one preferred pathway may require more detailed and continuous surveillance. For these cases it is not too difficult to relate the conditions at the relevant point of exposure to the conditions at the point of discharge; a reasonable approach to an engineering discharge limit is then achievable.

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In other cases, several pathways may have nearly equal significance to different end receptors. Characteristically, the limiting system may then change with relatively small changes in composition of the waste. Such would be the case for a combined discharge of Na^{24} and P^{32} to a river. The "point of exposure" could be varied both as to time and place. It could be made to apply to individuals swimming in the river (Na^{24}), to large fish, for example the sturgeon (Na^{24}), to fishermen catching small fish (P^{32}), or to hunters taking waterfowl (P^{32}), by varying the concentrations of the two isotopes and the access locations of the four animal types. Furthermore, the limiting "point of exposure" could vary with the seasons for fixed concentrations. In yet other cases, several pathways could all unite in the same ultimate receptor, the contributions from each pathway varying with the seasons. Some of these varying pathways of particular significance to waste disposal at Hanford are described in Part 2 et seq.

At this stage, it will be sufficient to demonstrate the main transfer mechanisms in note form:

Case 1.— Radioactive Wastes in Rivers, Streams, or Lakes

1.1- Hazards related to use as drinking water

1.1.1 Irradiation of gastrointestinal tract of man or animals.

- -.2 Transfer of radioactive materials to blood stream and irradiation of blood.

- -.3 Transfer to critical organ of deposition (e.g. liver, kidney, bone, or thyroid) as determined by the specific radioelement and its chemical and physical form.

- -.4 Irradiation of reproductive organs from materials passing through the body, deposited in specific organs, or in rare cases deposited in the reproductive organs.

1.1.5 Retention of filtrable radioactive materials in water purification systems, leading to external irradiation.

- -.6 Retention of non-filtrable materials in ion-exchange systems (e.g., water softeners) or in flocculent deposits in municipal water systems.

- -.7 Scaly deposits of radioactive materials from boiling hard water.

- -.8 Uptake by foods washed or cooked in domestic water.

1.2- Hazards arising from immersion in the water

1.2.1 External irradiation from the ambient water while swimming.

- -.2 Additional irradiation from materials attached to mud, rocks, aquatic plants; or, in fact, any surface.

- -.3 Irradiation from materials deposited on the skin or clothing.

- -.4 Swallowing of water, equivalent to Case 1.1.-

- -.5 Irradiation similar to Cases 1.2.1 and 1.2.2 from use of sanitary water.

1.3- Hazards arising from close approach to the water

1.3.1 External irradiation from river, lake, or reservoir.

- -.2 External irradiation from mud, sand, or plants near the shore.

- -.3 External irradiation from water while boating, water skiing, fishermen wading, etc.

- -.4 Augmentation of 1.3.3 by radioactive deposits on the boat structure, fishing gear, etc.

- -.5 Inhalation of contaminated spray. (→Case 2.--)*

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* Symbols such as (→Case 2.--) mean that this case leads also to some of the consequences of another case, such as Case 2.-- in this example.

1.3.6 Deposition of spray on skin or clothing. (→ Case 2.--)

- -.7 Similar situations from domestic use of sanitary water.

1.4.- Hazards arising from irrigation

1.4.1 Irradiation from water in canals or ditches.

- -.2 Retention of radioactive material in the surfaces of canals or ditches, persisting in the absence of irrigation water.

- -.3 Atmospheric pickup of radioactive materials from these cases or from irrigated soils. (→ Case 2.--)

- -.4 Direct spraying of plant surfaces with contaminated water by overhead irrigation.

- -.5 Immediate uptake of radioactive materials by plant roots from the water or delayed uptake following retention in the soil, in either case leading to incorporation in the plant structure; this can be followed by direct irradiation from the crops, or ingestion of the radioactive material by man or range animals.

- -.6 Further transfer and dissemination of radioactive materials by birds or insects.

- -.7 Transfer to the atmosphere by burning of crops.

- -.8 Radioactive milk, eggs, or meat from range animals or poultry fed as in 1.4.5.

1.5.- Hazards arising from biological chains in the river

1.5.1 Pronounced uptake of radioactive materials by attached algae or by plankton (free floating organisms).

- -.2 Transfer to snails, insect larvae, etc. directly or from 1.5.1

- -.3 Transfer to higher forms including edible fish.

- -.4 Ingestion of fish by man.

- -.5 Transfer to aquatic plants.

- -.6 Transfer to waterfowl from aquatic plants, insects, etc.

1.5.7 Ingestion of contaminated fowl by man.

- -.8 More exotic food chains such as ingestion of contaminated insects by frogs and the frog's legs by man.

- -.9 Mass emergence of contaminated ephemeral insects, their infestation of dwellings, and consumption by birds.

- -.10 Secondary contamination of bird nests and the like from excreta.

- -.11 Absorption of radioisotopes into fish eggs.

- -.12 Perpetual recycling of isotopes through biological chains through death and decomposition of organisms.

In any of these transfers, the transmitter as well as the receptor will receive radiation exposure.

1.6.- Hazards related to industrial processes

1.6.1 Direct irradiation equivalent to 1.3.-

- -.2 Preferential adsorption of radioactive materials on food products during washing.

1.7.- Hazards related to sewage disposal

1.7.1 Accumulation of radioactive materials in drains later requiring service.

- -.2 Accumulation in sludge and filter beds.

- -.3 Accumulation in bacteria, affecting their function.

- -.4 Concentration and return to agriculture via fertilizers.

Case 2.- - Radioactive Wastes Released to the Atmosphere

2.1.- Hazards related to inhalation

2.1.1 Direct irradiation of the lung due to radioactivity of the contained air.

- -.2 Solubility of radioactive gases or vapors and transfer to the blood and the specific organs of deposition. (→1.1.2 and 1.1.3)

2.1.3 Deposition of radioactive particles on the surfaces of lung, bronchi, and upper respiratory passages, giving intense local irradiation.

- -.4 Solubility of some of these particles. (→2.1.2)

- -.5 Transfer of particles to lymph nodes of the lung.

- -.6 Removal of particles by ciliary action, leading to ingestion. (→1.1.-)

2.2.-Hazards from radioactive material in ambient air

2.2.1 External irradiation of the human or animal body, or plant tissue.

2.3.-Hazards related to deposition of materials

Note: Although the process is customarily referred to as fallout, the principal effect is the bringing of the radioactive material close to the ground by turbulent diffusion of the atmosphere, and its physical impaction against various surfaces. In industrial releases, there are only rare cases in which there is a physical fall of the materials under gravity. Deposition of airborne material is of course enhanced, sometimes quite strongly, by precipitation which may either be formed around particles or wash them out of the atmosphere as the raindrops or snowflakes fall.

2.3.1 Irradiation from deposition on skin of man or animals, or on clothing.

- -.2 Transfer of radioactive materials from other surfaces to skin.

- -.3 Secondary transfers to food, cigarettes, etc., leading to ingestion.

- -.4 Deposition on crops, followed by transfer to man or animals.

2.3.5 Secondary transfer of the radioactive materials through use of products such as milk, eggs, and meat.

- -.6 Deposition on soil. (→1.4.5)

- -.7 Deposition on bodies of water. (→Case 1.- -)

- -.8 External irradiation from deposited materials.

- -.9 Transfer from one animal to a predator animal.

- -.10 Washing by rain or floods into water systems. (→Case 1.- -)

- -.11 Real fallout of individual highly active particles or flakes onto hair or skin.

Case 3.-.- Ground Disposal of Liquid Wastes

3.1.- Hazards at the disposal site

3.1.1 If to surface ponds, the hazards are repetitious of many parts of Case 1.- -, particularly 1.3.- (The ponds may be particularly attractive to wildfowl.)

- -.2 If contaminated surface is dried, transfer of radioactive materials to atmosphere by wind pickup of soil. (→Case 2.- -)

- -.3 Direct irradiation from a source just below ground.

- -.4 Seepage of radioactive gases or vapors from below ground.

- -.5 Flash floods removing superficial active materials. (→ Case 1.- -)

3.2.- Hazards resulting from underground travel

3.2.1 Return of radioactive material to plant life from deep rooted plants.

- -.2 Subsequent leaching out of radioactive material apparently fixed in soil.

- -.3 Chemical reactions releasing materials as incompatible wastes are later added.

3.2.4 Slow fall of wastes under gravity when apparently safely separated from the ground water.

--.5 Entry into ground water and subsequent recovery from wells. (→ Case 1.--)

--.6 Underground travel to springs or rivers. (→ Case 1.--)

Actual Magnitude of Exposures In the Environs of the Hanford Plant

Although practically all of the pathways presented in note form in the preceding section have been investigated to some degree, experience has shown that several of them predominate to such an extent that relatively few require either continuous or intensive observation.

Due to the specific nature of many of the measurements, the results reported below represent exposures above the background contributed by natural sources. This permits direct comparison with the various recommendations for control, since these recommendations are specifically for man-made sources of radiation. For radiation exposures to people outside of the project boundary, the recommended control limits may be broadly summarized as annual exposures of 500 mrem for whole body irradiation, and 1500 mrem to most soft tissues from internally-deposited emitters. Throughout this section doses given in rem or mrem mean RBE doses within the meaning of our glossary; endless repetition of RBE dose is thus avoided.

Swimming in the Columbia River accounts for about 0.02 mrem per hour and would thus make only a modest addition to the total exposure. An enthusiastic swimmer (2 hours every day) could accumulate an annual dose from this source of about 15 mrem. Boating for 4 hours every day would give a like contribution. The maximum exposure of this type is probably received by persons whose occupation requires that they work on the river in the vicinity of Richland. Measurements indicate that an annual dose of about 40 mrem would be received by such persons who spend one-third of their time working in this manner.

The drinking of water from the Pasco or Kennewick systems provides a dose of perhaps 50 to 100 mrem per year to the gastrointestinal tract, and less than one-half of this to other organs of interest. Consistent use of unfiltered river water could lead to doses up to 200-300 mrem per year.

The food chains in the river leading to accumulation principally of P^{32} in both game fish and resident waterfowl lead to an averaged annual dose to human bone on the order of 10 mrem. Doses ten to twenty times these values are conceivable for individuals who harvest and eat an unusually large number of fish and ducks. As will be shown in Part 2, the sharp seasonal variations in the radioactive content of these life forms make it prudent to maintain thorough surveillance of these values at all times.

In the releases to the atmosphere, the critical system is through the uptake of I^{131} from vegetation. This system receives continuous examination to provide full assurance of the protection of range animals as well as humans. Careful experimentation has shown that the accumulation of I^{131} in the thyroid gland of sheep (the principal grazing animal of this area) will be appreciably below the amount which causes damage when the average concentration of I^{131} on the vegetation is 1×10^{-5} μc/gm. Although some patches of vegetation within the boundaries of the reservation may, at times, have I^{131} concentrations above this level, there is no evidence that the populations of wild animals which eat it are affected by the higher exposure. In all a sum of approximately $3,000,000 has been spent on research studies and field investigations on this one problem. The transfer to man through a similar channel is lower; it accounts for about 10-20 mrem per year.

The local vegetation contains trace quantities of many other fission products, which would irradiate the gastrointestinal tract upon ingestion, with lesser exposures to bone and reproductive organs. It can be shown that most of these exposures in the Hanford environs are due to fallout from nuclear

weapons, rather than to local sources. Estimates of exposure based on extensive analyses of general vegetation samples indicate an average exposure to the gastrointestinal tract of less than 50 mrem per year. Actual analyses of produce from the region would indicate that exposure to the gastrointestinal tract would not exceed 100 mrem per year even for the individual of extraordinary eating habits.

In addition, the general radioactive contamination of the ground contributes an annual dose of about 10 mrem. It will be recalled that the annual dose from natural background radiation is in the range of 140-180 mrem.

The ground disposal pathways add nothing to local exposure, because the "point of exposure" has not been reached.

Man is one of the receptors who receives small contributions from many sources. A reliable appraisal of the actual doses received by large numbers of people is difficult to obtain. It is influenced by wide variations in diet, individual metabolism, personal habits, occupancy factors - in fact, just about the whole interaction of individual man with his environment. Also, the dose contributions from different sources are limited to different parts of the body including various internal organs or tissues. It follows that the simple picture of a national register for obtaining such doses, for example by film badges, is quite irrelevant.

We have given considerable attention to the integration of these doses in the Hanford environs. The frame of reference for these complex estimates is presented at the end of this part so as not to interrupt the discussion. It is, however, vitally important to a balanced appraisal of the data. Within this framework it appears that, in man:

1. According to present calculations, the gastrointestinal tract of man is the organ system receiving the highest dose, followed by bone and the reproductive organs, with whole body, lung, and thyroid doses relatively insignificant.

The GI tract dose is made up of 4 or 5 factors derived from radioactive material released to the river, and two from releases to the atmosphere.

The summation of these 6 or 7 components tends to compound the errors inherent in each separate estimate. It is probably fair to accept that the representative annual dose in the environs from other than natural sources is not less than 50 mrem and not more than 200 mrem. The accepted annual limit for GI-tract dose is 1500 mrem in non-controlled areas.

If all 6 or 7 contributing factors were maximized in the unfavorable direction for one individual, an annual dose of 600 to 700 mrem would be estimated. The comparative uncertainty of some of these dose calculations has been emphasized. It may thus be supposed that the safety margin for this case is not a comfortable one. Realistically however, the permissible annual GI tract dose within the controlled area is 15 rem (ie: 15,000 mrem) and this is believed to be without deleterious effect. Since the GI-tract can hardly be supposed to be aware of this differentiation between controlled and non-controlled areas, there is, in fact, a wholly adequate safety margin.

2. In the case of bone irradiation, the permissible annual limit is expressed in a rather peculiar way not quite equivalent to 500 mrem. In the same terms, the probable annual bone dose of representative individuals in the Hanford environs may be expressed as not less than 3 per cent of the limit, and not more than 10 per cent of the limit. For the quite exceptional individual this may rise to an estimated 50 per cent of the limit. We would subscribe to the view that the bone limit commands more respect than the GI tract limit. Thus, although again the occupational limit is formally ten times higher than that given here

we would be concerned if such levels were created in bone, and especially by long-lived emitters such as Sr^{90}. In the Hanford case, the principal source of bone deposition is P^{32}. Of the various radioisotopes whose metabolism has been carefully studied at Hanford, P^{32} is the one exception in which we expect that our researches may lead to an increase in the permissible intake. This would reduce the apparent bone dose as a percent of the limit. There is the further practical point that a given deposit of P^{32} in bone can be reduced by a factor of 10 within 7 weeks by simple radioactive decay. This effect would be neglibible in the case of Sr^{90}.

3. The dose to the reproductive organs, with present technology is not easily estimated. Within the specified frame of reference it is perhaps safe to say that reproductive organs of most people in the vicinity of the Hanford plant probably receive an annual dose of not less than 20 mrem and not more than 50 mrem, from atomic energy sources, including fallout. It is virtually certain that no individual receives an incremental genetic dose in excess of 200 mrem per year from Hanford waste disposals plus fallout.

   In the consideration of genetic effect, it is the average dose integrated over the whole population segment that is of interest. From Hanford waste disposal practices this average dose is evidently of minimal significance.

In summary, for man in the Hanford environs, the GI-tract dose is generally the highest organ dose. Expressed as percentage of the corresponding permissible limits, GI-tract dose, bone dose, and reproductive-organ dose rank about equally. The estimated representative contribution in each case

is in the range of 3 per cent to 15 per cent of the limit. The corresponding maxima for exceptional cases where unusual amounts of local fish and leafy vegetables are eaten seem to be contained within the general range of say 40 per cent to 60 per cent of the limits.

All these numbers are derived by methods of calculation that are believed to tend to overstate the magnitude of the exposure. The possibility of understating some portion of a component contribution cannot be entirely excluded. Therefore, critical and definitive investigations are continuing on the components that either make the major calculated contributions or contain greater uncertainties.

Hanford Effort on Waste Disposal

The total man-power involved in waste disposal operations, plant and environs sampling and evaluation of the data at Hanford is currently about 155 man-years per year. In addition are about 100 man-years per year for research and development on radioactive waste disposal.

In the environmental monitoring program about 40,000 samples per year are collected, with about 75,000 analyses performed on the samples. An additional 40,000 field measurements are recorded from monitoring equipment such as detachable ionization chambers, fixed and portable survey instruments, and radiation-sensitive film.

In the total environmental program, gaseous waste discharged to the atmosphere is monitored by collecting 10,000 samples per year and by continuous monitoring at the point of discharge. Liquid wastes discharged to surface waters and the ground are monitored in and around the plant by collecting 17,000 samples and making 42,000 specific measurements per year. Samples are taken of sanitary, river, swamp, ditch, and underground waters. The deposition of radioactive materials on vegetation and ground is monitored

by collecting 11,000 samples and making 17,000 field measurements per year. Crop sampling is conducted during the growing season. Other biological monitoring includes 500 birds, 700 rabbits and 1,200 samples of aquatic organisms collected per year. The distance travelled in obtaining samples and measurements as far as 250 miles from the project, is about 350,000 miles per year. An additional 12,000 miles per year are flown in airplanes in the conduct of routine surveys.

Since 1943, about 600 test wells have been drilled to depths of 25 to 600 feet in the Hanford reservation. The total footage drilled is about 120,000 feet, or roughly 23 miles.

Eighteen people are engaged in collecting samples and making field measurements. Another twenty people are engaged in laboratory analyses, counting, and preparing results for computing machine processing. Evaluation and reporting of results involves eight people.

The capital costs of all Hanford facilities or systems involved in the collection, handling, processing, monitoring, and control of gaseous, liquid, and solid waste effluents are approximately $114,000,000.

The related annual operating and maintenance costs are about $2,000,000. Related research and development activities are conducted at a level of $1,000,000 annually. These figures are introduced to define the order of magnitude of the effort. In detail, arbitrary decisions have to be made on the division of costs of facilities and services that also have other functions.

Professional Relationships

Extensive representation on various national committees concerned with radiation protection is maintained by the Hanford personnel. Obviously these relationships are determined by individual professional stature and

are not organizational representation. Nevertheless, they serve in addition to transmit the Hanford studies to these committees and to bring to this site the benefits of related studies by others. Of the principal committees significantly related to waste disposal problems, there are seven representatives on the executive, main, or sub-committees of the National Committee on Radiation Protection and Measurement, and seven on the committees or sub-committees of the National Academy of Sciences - National Research Council study of the biological effects of atomic radiation.

There are conventional professional contacts with contractor and AEC personnel at other AEC sites. Through approved channels, there has been some stimulating exchange of information with United Kingdom and Canadian atomic energy establishments.

Locally, there is a mutually beneficial arrangement with the University of Washington and Washington State College; notable phases of this include contacts with authorities in various fields of the life sciences in connection with Hanford's biological experimentation. Exchange seminars are also being established with these and other universities of the Pacific Northwest.

From the original choice of the Hanford site to the present time, the United States Geological Survey has assisted the Hanford Works in geological exploration, well drilling, and hydrological investigation and research. Agencies which have reviewed Hanford disposal practices include the Safety and Industrial Health Advisory Board in 1948, the U. S. Public Health Service which conducted a radiological and biological survey of the Columbia River in the period 1952-1954, and the Columbia River Advisory Group.

The Columbia River Advisory Group (CRAG) was organized in 1949 to advise the Atomic Energy Commission on the control of effluents released to the local environs, and conversely, on the potential influence of outside activities on

the Hanford project. The members of CRAG are pollution experts representing the States of Oregon and Washington with a chairman representing the U. S. Public Health Service.

In addition, Hanford management maintains appropriate contacts with other public officials and agencies, such as the Corps of Engineers, Bureau of Reclamation, fish and wild-life authorities, U. S. Department of Agriculture, U. S. Weather Bureau, local public health and police officials, and public power agencies.

Dissemination of Information

Some concept of the large volume of reports and articles which have been prepared on Hanford waste disposal, including research and development work on various aspects of the problem and on measurements of exposure in the environment, can be gained from the bibliography presented as Part 7. This listing includes over 150 documents, approximately half of which are unclassified and all of which have been made available through the Atomic Energy Commission's Civilian Application Program. In addition to approximately 40 articles which have been published in journals of technical and professional societies, about 200 oral presentations have been made to appropriate scientific groups.

As early as 1951 and 1953, special exhibits were prepared concerning, (1) the effects of I^{131} on sheep and (2) the biological aspects associated with discharge of reactor effluent to the Columbia River. These exhibits were first displayed at national meetings of the Radiological Society of North America, but have since been shown to a variety of professional, stockmen's, and sportsmen's groups in the Pacific Northwest.

Of special significance are the detailed reports on waste disposal experience at Hanford which have been included in the International Conferences on the Peaceful Uses of Atomic Energy held in Geneva, Switzerland. Seven papers appear in the Proceedings of the 1955 Conference and ten will be included in the Proceedings of the 1958 Conference.

Not included in the bibliography are a number of informal talks given to service clubs in most of the neighboring cities, radio broadcasts designed to inform the public on the Hanford operation, and conducted tours which have shown research work in progress and disposal facilities to several hundreds of interested citizens of the region.

In addition to over 72 releases printed in the local press which have referred to work associated with waste disposal, 55 feature articles dealing with various aspects of waste disposal and the environment have appeared in leading newspapers and magazines of the nation.

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FRAME OF REFERENCE FOR CALCULATION OF RADIATION EXPOSURE FROM ENVIRONMENTAL RELEASES OF RADIOACTIVE MATERIAL

Experience at such installations as Hanford has shown that the relevant total dose to any life form is likely to be derived from a large number of pathways.

As an example, the local contribution to gastrointestinal-tract dose requires consideration of 6 or 7 significant mechanisms, viz:

Derived from Radioactivity in the River

1. Swimming
2. Boating, wading or close access to the river
3. Drinking river water
4. Eating Columbia River sports fish
5. Eating resident waterfowl (or this could be grouped with the fish as one source)

Derived from Radioactivity in the Atmosphere

6. External radiation from material on the ground
7. Eating vegetation

Many other pathways, such as inhalation of radioactive particles, must also be examined and the decision reached that the sum of these contributions is small compared with the listed ones. Then each mechanism must be examined in detail. For example, in #3 - drinking of river water, the possible contribution of some 60 radioisotopes each with individual characteristics has to be integrated. A practical decision must be reached that some smaller number, say 10 to 20 will encompass the significant contribution.

In almost all cases, the dose contribution either from a principal category or from the subcomponents of a category must be estimated by quite indirect means. In the chosen example, the dose rates to the body from categories 1, 2, and 6 can be reasonably measured. Translation to integrated doses to the organ then requires only data on the depth penetration of the

radiation and assumptions on the occupancy factors. Competent observers would thus derive about the same results for these categories.

The situation is notably different for categories 3, 4, 5, and 6, and for the numerous minor pathways here excluded.

Plausible general principles to be applied to studies of this kind are:

(1) Where a particular component clearly makes a very low contribution to the over-all dose, a considerable margin of error in the estimate is quite inoffensive; as an example, if a particular chain of phenomena is estimated to produce exposure of 0.1% of a particular permissible limit, it is immaterial whether the actual exposure is 0.01% or 1%. There may be some academic interest in developing fuller understanding for other applications.

More cogently, there may be three practical needs for more precise information:

a. If this particular chain and many more like it (say 50) may all be underestimated for related reasons, more reliable analysis is needed. This is clear because the calculated sum is 50 x 0.1% or 5% of limits, whereas the real sum might be say 50 x 1% or 50% of limits, which would be highly significant.

b. If the feasible uncertainty of the calculation is very large, say 0.001% to 10% of limits, the provisional estimate of 0.1% of limits is not safe.

c. If the present origin of the chain is intended to be increased by a factor of 10, a more critical analysis may be needed. This follows because if there are several chains involved the former estimate of 0.1% for the single chain goes up to 1%,

but there is no assurance that it may not be near the new upper limit of 10%.

(2) Because of these uncertainties, it is conventional for radiological physicists to apply assumptions in their calculations that tend to maximize the apparent exposure. A peculiar and disturbing feature arises from this, as follows:

In many real cases, the chain of phenomena may contain five factors, or schematically,

True dose $D = a \times b \times c \times d \times e$

A competent qualified expert, A, examines the case, and concludes that the appropriate values of a, b, c, d, e are

a_1, b_1, c_1, d_1, e_1, where

$a_1 = 3a$, $b_1 = 3b$ and so on

His estimated dose $DA = 243\ D$.

(3) An equally competent qualified expert B concludes that the appropriate values are a_2, b_2, etc.

where $a_2 = 1.1a$ and so on.

Because of the general uncertainties of the case he assigns a final safety factor of 5.

His estimated dose $DB = 8.05\ D$.

(4) Mr. A's value is 30 times Mr. B's.

(We have shown each calculation to 3 significant figures, which is also characteristic of some work in this field when even the first significant figure is not assured. This tends to give a misleading aura of reliability.)

The artificial example given is a modest one; there are cases on record today in which well-respected observers derive answers from the same partial data that differ by such a factor.

The observation of such differences by an enquiring public is not calculated to lead to a full measure of confidence in either the skill or integrity of the scientists involved. An important point to be made is that such differences will naturally arise by multiplication of successive safety factors in these sequential reasoning problems. They reflect nothing more than variance in the degree of conservatism in qualified individuals. They do not necessarily imply bias or prejudice by either observer to demonstrate a plausible unfavorable or favorable answer. A sympathetic understanding of this feature by an informed public can contribute substantially to keeping radiation hazards in proper perspective.

(3) Whenever possible, the relevant measurement (D of the artificial example) should be made. As a next choice, a measurement that effectively measures the partial product (a x b x c x d) should be made. This leaves only the uncertainties in the factor e and the relatively small experimental error in the partial product. As a next choice, the partial product (a x b x c) should be measured and so on.

(4) Using the posture of the radiological physicist we have perhaps implied that determination of the true dose D is the principal objective. This assumes that the biological consequences of a given dose are known. This is by no means the case especially for the complex patterns of internal deposition of radioactive isotopes resulting from waste disposal.

Direct biological investigation is therefore a necessity, and it must be successfully integrated with the preceding steps from other disciplines.

(5) With the best available data on hand, the remainder of the calculation is to be completed with assumptions maximizing the apparent dose or minimizing the apparent permissible limit, as the case may be.

(6) In any practical application in which responsible investigators derive divergent estimates, and the more pessimistic one shows a potentially hazardous situation, the rational approach is to go to the laboratory and field investigations for better data. As indicated in the reference to I^{131} at Hanford this can be an expensive and time-consuming obligation. Using this case as an example again, it can be systematized to an 8 factor chain between emission from a stack and biological effect of I^{131} on lambs at a particular location. In the absence of our local research on this, the product of the sequential uncertainty factors could easily reach the range of 5,000 to 10,000. The price paid to reduce this uncertainty factor to a manageable size is very small compared with the cost otherwise required to reduce the emission by orders of magnitude.

(7) The relative merits of laboratory and field studies require specific examination for each case. In general, the laboratory has these advantages:

a. With adequate foresight, the key experiments can be done before the disposal process is in place.

b. The conditions are controlled; damage levels can be found under conditions intolerable in the environment; the effects on successive generations can be followed.

c. Small animals can be substituted at the exploratory stage; expensive large animal confirmation is needed only in certain key problems.

Field investigations have these merits:

a. They unequivocally demonstrate the real conditions.

b. They suggest leads for more exacting laboratory work.

Since the food webs in nature are complex, no environmental study is complete at the laboratory stage, as a critical intermediate organism may be omitted.

(8) Translation of results from one environment to another is not always possible. Regional, racial, or national dietary and other characteristics may have a marked influence.

(9) Finally, the enthusiastic observer should not overlook the possibility that the combined cost of all studies of a hazard problem may exceed the cost of eliminating the waste release by an engineering improvement.

PART 2

RELEASE OF REACTOR COOLING WATER TO THE COLUMBIA RIVER

R. F. Foster

Radioactive wastes generated by the operation of the reactors are almost exclusively those which are associated with the cooling water. After passing through the reactor and picking up heat and radioactivity the spent coolant is referred to as reactor effluent. Since the kinds and amounts of radioactive materials produced in the effluent are partially dependent upon the characteristics of the cooling water fed into the reactor, close control of water quality is maintained. Such control may be considered as the primary phase in management of the reactor effluent. The flow of water through a typical reactor area is illustrated in Figure 1.

Water Purification and Treatment

The Columbia River water used to cool the reactors is processed in water treatment plants similar to those found in many cities. The raw river water always contains finely suspended matter although the amount varies greatly from time to time. This is coagulated with the use of alum. Some of the coagulated matter settles and the remainder is removed by filtration through beds of coal and sand like those shown in Figure 2. During the periods of high turbidity an organic material is used to assist the alum in the clarification process. Although most of the Hanford plants are processing over twice as much water as would be considered normal for their size, water of exceptional clarity is produced. The chemicals used for clarification cost about two dollars for each one million gallons of water. Many comparable plants spend from two to five times as much for clarification. Regulation of the pH and addition of sodium dichromate complete the treatment.

(For figure 1, referred to above, see exhibit 6, p. 189.)

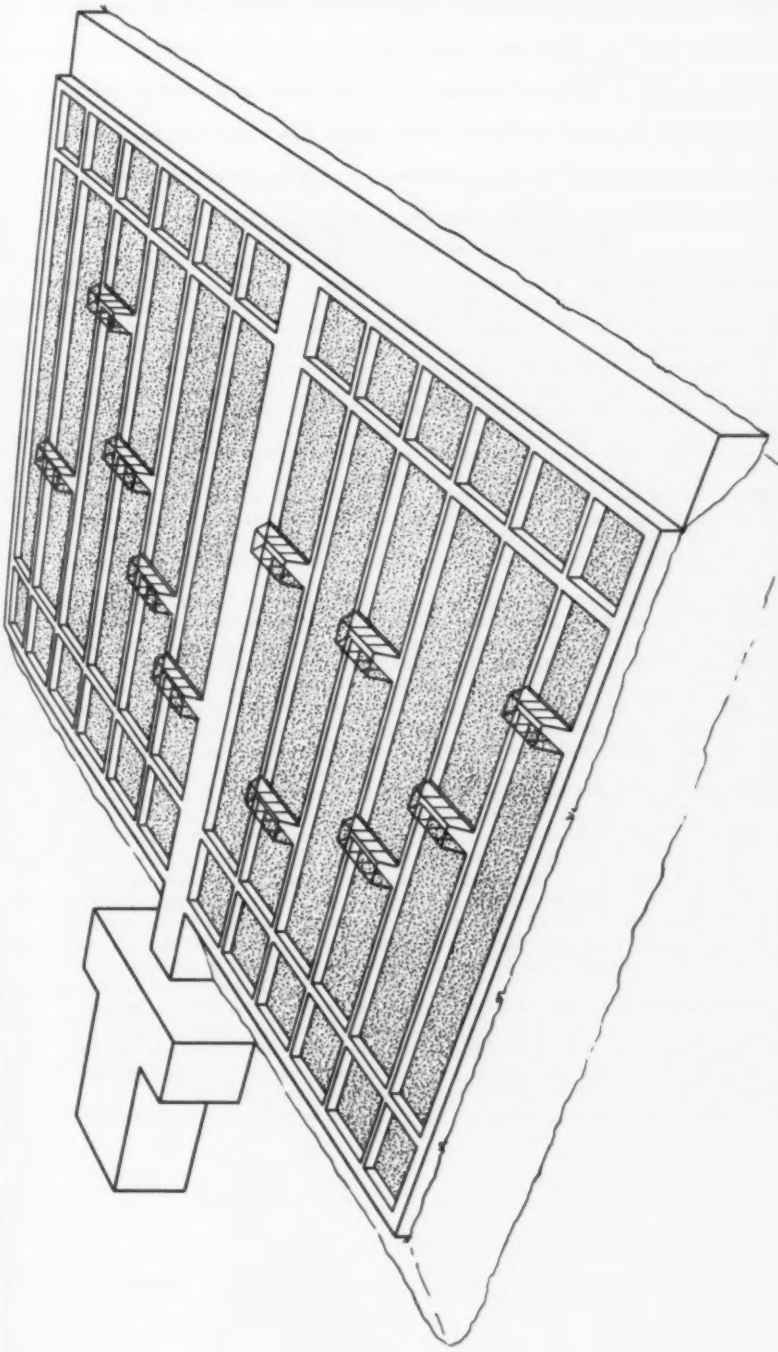


FIGURE 2
FILTER BEDS USED TO CLARIFY COLUMBIA RIVER WATER FOR PROCESS USE

Control of the filtration process is now based on the clarity of the filtered water which is measured at the beginning and end of each filter cycle by means of a special light-scattering microphotometer developed at Hanford for this purpose. Frequent analyses of the water are also made to assure that the pH and concentration of dichromate are within specified limits.

Origin and Characteristics of Radioactive Materials in the Reactor Effluent

Radioactive isotopes are formed in the reactor by the neutron bombardment of material present in the neutron flux. They are formed by one of several nuclear reactions which may either produce radioisotopes of the same chemical element as the parent material or radioisotopes of a different chemical element. Some of the radioisotopes which appear in the effluent are formed from parent material present in the cooling water fed to the reactor, some result from corrosion of the structural materials of the reactor.

Sodium can be used to illustrate the activation process. This element is a common constituent of the cooling water fed to the reactor and a substantial part of the radioactive sodium-24 measured in the effluent originates from this source. Sodium-24 can also be formed from aluminum, however, and since aluminum is used for jackets of the fuel elements and for some of the reactor piping, a substantial part of the radiosodium comes from this source. Still another portion of the sodium-24 is formed from a magnesium isotope.

In some cases it is not necessary for the parent element to be present in great quantity. In the case of radiophosphorus (P^{32}), for example, all of the isotope present in the effluent could be formed from such a minute amount of phosphate ion in the cooling water that the concentration could not be accurately measured by conventional chemical methods.

The exact source of many of the important effluent isotopes is as yet unknown. One of the most promising mechanisms for control of the kinds and amounts of radioactive materials entering the Columbia River depends upon such information, however, since even modest reductions in the concentrations of a very few elements present in the coolant might substantially reduce the formation of the more significant isotopes. We do know that the bulk of the radioactive materials which appear in the effluent are activation products of stable isotopes present in the water, in the film which forms on the reactor tubes and fuel elements, or in the metals of which the reactor components are constructed. Some fission products also occur because of natural uranium dissolved in the Columbia River water and as a consequence of occasional ruptures of the aluminum jackets of the fuel elements.

Reactor fuel elements are cylinders of uranium encased in aluminum cans as shown in Figure 3. The aluminum can is necessary because uranium oxidizes rapidly in hot water. Occasionally, through erosion, corrosion or stress, the aluminum can develops a hole through which the thermally hot water in the reactor can reach the uranium. This is commonly called a rupture. When a rupture occurs, oxidation of the uranium takes place, releasing fission products to the cooling water. As many as 170 different isotopes are possible. These fission products contain a much higher percentage of long half-lived radioisotopes than the fission products encountered immediately following an atomic bomb burst. During the relatively long irradiation time of the fuel element the short-lived radioisotopes decay, while the long-lived isotopes build up.

Efforts are being made to improve the integrity of fuel element jackets, but meanwhile ruptures occur spontaneously and irregularly. The reactor is shut down as soon as a rupture is known to exist and the fuel column

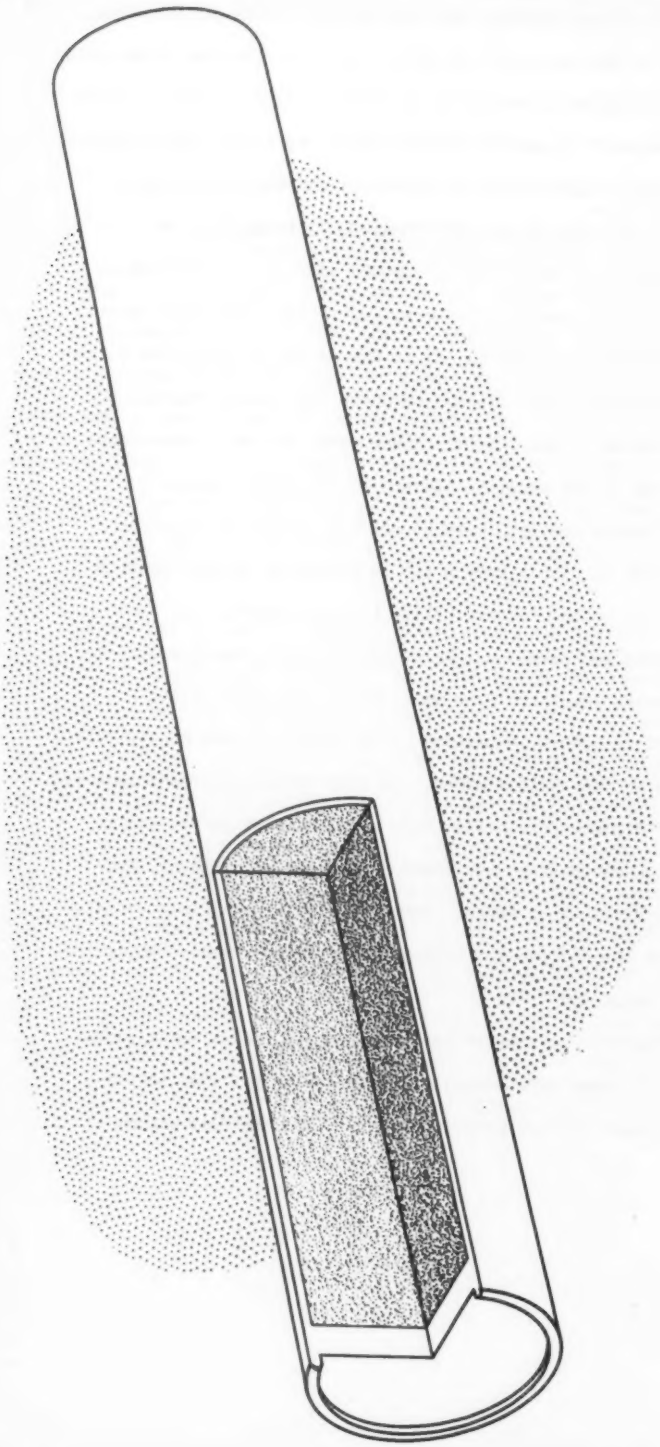


FIGURE 3
TYPICAL FUEL ELEMENT USED IN THE HANFORD REACTORS

containing the rupture is discharged. The time between initial failure and discharge may be several hours, and meanwhile the rupture releases fission products to the coolant stream.

The total quantity of fission products released varies greatly from one rupture incident to the next and the instruments used to monitor the effluent on a routine basis are not yet capable of measuring the exact amounts. Although significant amounts of fission products have been released in this manner, ostensibly to the river, pulses in the activity density in the river downstream are also difficult to measure. It is estimated that ruptures which occurred during 1958 may have contributed about five per cent of the radiation exposure received by people in the Pasco and Kennewick area from the drinking of Columbia River water. The most effective control measure is rapid detection of a failure and subsequent shut down of the reactor.

The water treatment process used at Hanford was developed to produce water with very low turbidity because colloidal material carried into the reactor with the process water forms a film on the aluminum surfaces of the tubes and fuel elements. When this film builds up to the point where its interference with cooling water flow and heat transfer outweighs its benefit of inhibiting corrosion, it is removed by "purging". "Purging" is accomplished by adding finely powdered diatomaceous earth to the process water for short periods of time. The abrasive action of these fine particles wears away the film without measurable erosion of the aluminum surfaces.

Purge frequency varies widely between reactors and between seasons. For instance, in 1958 one reactor was not purged at all, whereas, at the other extreme, another reactor was purged at least thirty times. Altogether there were about 50 purges among the eight reactors during 1958. The concentration of several isotopes in the reactor effluent increases preferentially

TYPICAL FUEL ELEMENT USED IN THE HANFORD REACTORS

during purges; Fe^{59}, Zn^{65}, P^{32}, Cu^{64} and As^{76} in descending order. Greater than 90 per cent of the increase in the reactor effluent contamination that occurs during purges is closely associated with the solids in the effluent water. The quantity of radioactive material released in the reactor effluent during purges varies widely. Such variations strongly suggest that the quantity is primarily a function of the amount of film available for removal in the reactor which in turn is a function of the time interval between purges and water film-forming qualities.

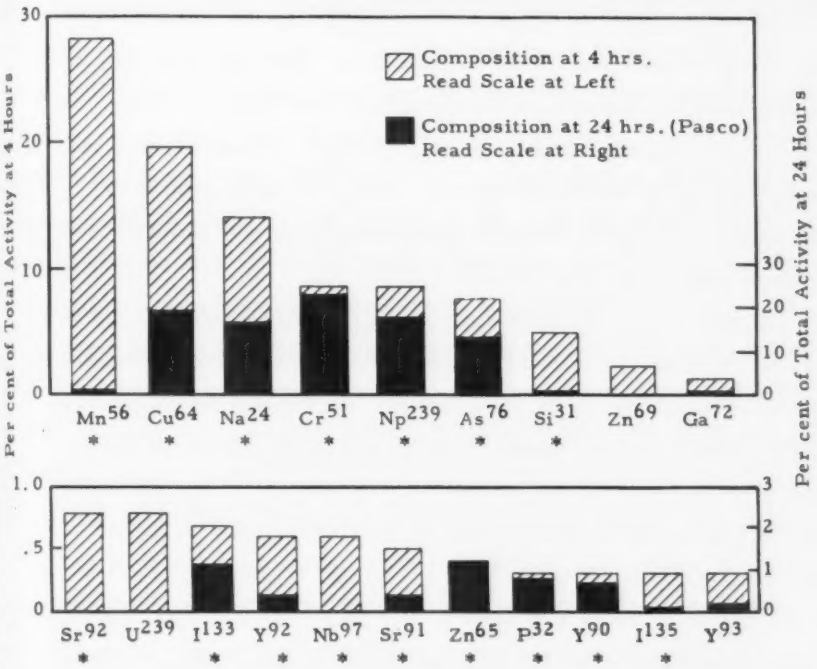
When a reactor is purged, the gross quantity of radioactive materials in its effluent is increased by as much as a factor of six during the short time interval that the purge is in progress. This time interval is very short especially in comparison to the total operating time of the reactors but, as an added control purges are spaced so that no more than one reactor is purged during a single day. In 1958 the amount of radioactivity released to the river was probably not increased by more than 0.5 per cent by purges.

More than 60 radioisotopes have been identified in the effluent and Figure 4 shows their relative abundance under normal circumstances four hours after irradiation. Since the rate of radioactive decay is different for each isotope, the relative abundance of the various isotopes is constantly changing. By the time the effluent has traveled to the vicinity of Pasco, a trip which requires about one day, the effluent composition might be expected to resemble that indicated by the shaded portions of the bars in Figure 4. Of the total number of isotopes detected in the effluent, 24 are of sufficient interest that their concentration is measured on a routine basis.

Per cent of Total Activity at 4 Hours

Retention and Disposal of the Effluent

Since the design of the Hanford reactors is based on a once-through type of cooling system, management of the radioactive waste must be based



In addition to the isotopes shown above, which contribute about 98 per cent of the activity 4 hours after irradiation, trace amounts of the following have also been found.

Eu-152	Eu-157	I-131*	Y-91	Pr-145	Ca-137
Sm-153	Ba-140*	Ce-141	Fe-59*	Pm-151	Sr-85
W-187	Mo-99	Pr-142	Sr-89*	Co-60	U-238*
La-141	Sm-156	C-14	Mn-54	Pr-143	Pu-239*
Nd-149	Sc-46*	Nd-147	Zr-95	Ru-103	Ac-227
La-140	Cd-115	Ca-45*	Pm-149	Sc-47	Po-210*
I-132*	Ce-143	Ag-111	Eu-156	Sr-90*	

* Routine measurements are made on these isotopes.

FIGURE 4
ISOTOPIC COMPOSITION OF REACTOR EFFLUENT
4 HOURS AND 24 HOURS AFTER IRRADIATION

not only on the production of isotopes which appear in the effluent but also upon release of the wastes to the Columbia River. Some degree of control is accomplished by retention of the effluent in large basins or tanks. The basins of the original reactors were constructed of concrete but this has resulted in serious maintenance problems because of faults produced by temperature shock. Steel tanks are used for the three most recently constructed reactors.

Although the effluent is retained in the basins for only one to three hours before release, this is long enough for radioactive decay to reduce the gross radioactivity by 50 to 70 per cent. By the time the effluent has traveled about 35 miles, radioactive decay has further reduced the gross activity to less than ten per cent of the level when it entered the basins. Figure 5 illustrates the rate of decay of the gross activity in typical effluent.

The retention basins also serve another purpose in that they intercept effluent which has an unusually high radioactive content. Such situations arise from the occasional ruptures of fuel elements and from the periodic purging of the film which forms on the reactor tubes. A part of such waste is in the form of minute particles and some of these settle to the bottoms of the basins. About 40 to 70 per cent of the radioactive material from the purges apparently settles out in the retention basins. In some cases, effluent which contains purge material or rupture debris is diverted into trenches or "cribs", where it seeps into the ground. This provides both filtration and decontamination by ion exchange before the liquid reaches the river. No allowance for such removal is made in the computation of quantities of radioactive material released to the river, however.

(For figure 5, referred to above, see exhibit 7, p. 190.)

The condition of the retention basins for five out of the eight reactors at Hanford makes it impossible to direct effluent water laden with rupture debris to trenches or cribs. These are the five older basins which are constructed of concrete. Each of these basins is divided into halves by a concrete wall. Before power levels, and thus the temperature of the effluent, were appreciably increased, only one-half of the basin was used at a time. This allowed switching flow into an empty half-basin following a rupture. The basin which contained the rupture debris was then drained to the crib or open trench and the insoluble fission products were thus filtered out before the water entered the river. With higher temperatures and deterioration of the basins with time, use of only one-half the old basin is no longer possible because of the effect of thermal shock on the integrity of the basins.

After retention in the basins, the effluent enters the Columbia River via large pipes which are buried in the river bottom. The outlets of these pipes are from 200 to 1600 feet from the plant shore so that the effluent is immediately mixed with a large volume of water flowing in or near the main channel of the river. Figure 6 illustrates a typical installation. In the event that the tubes which extend into the river should become inoperable, there is provision for disposal of the effluent via emergency flumes which discharge to the shore line.

Based on present day construction costs, the effluent retention and disposal systems represent an investment of about two and one-half million dollars for each reactor unit.

Distribution of Radioactive Waste in the Columbia River in Relation to Water Use

The effluent discharged to the river immediately becomes distributed between the bottom and the surface because of the force with which it leaves the discharge pipes, its high temperature, the turbulence of the

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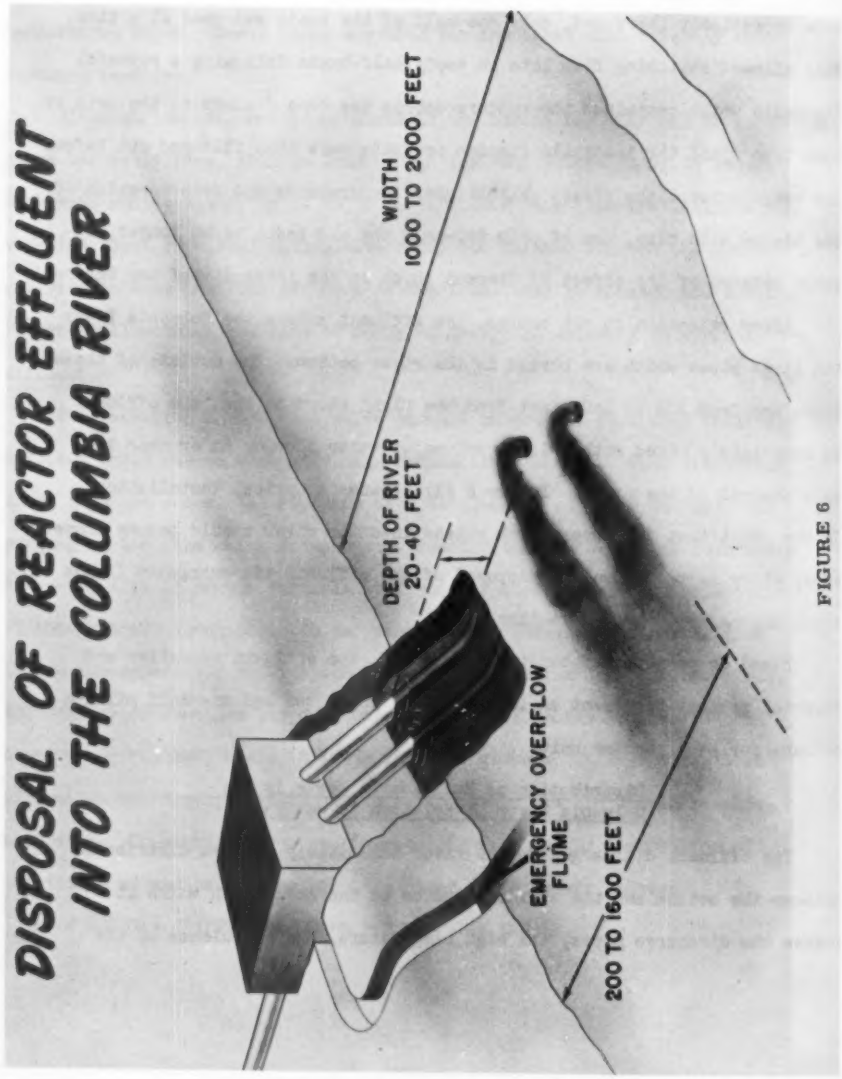
DISPOSAL OF REACTOR EFFLUENT INTO THE COLUMBIA RIVER
WIDTH 1000 TO 2000 FEET
DEPTH OF RIVER 20-40 FEET
EMERGENCY OVERFLOW FLUME
200 TO 1600 FEET

FIGURE 6

river, and the relatively shallow depth - 20 to 40 feet compared to a width of 1,000 to 2,000 feet. The effluent is slowly spread by turbulent diffusion as it is transported downstream. A typical dispersion pattern from one reactor is illustrated in Figure 7. The tendency for high concentrations of the effluent to remain in midstream is advantageous to downriver reactors since their shoreline intakes pick up a smaller fraction of the waste.

The first point of public use of river water below the reactors is at Ringold which is shown on Figure 1 of Part 1. This farming area is on the opposite side of the river and some eleven miles downstream from the last reactor. Dispersion of the effluent is incomplete in this region and thus the river water picked up from near shore by the small private irrigation systems contains less than average amounts of the effluent.

Even after traveling a distance of about 30 miles downstream to the vicinity of Richland the effluent is not quite uniformly mixed throughout the river. The Yakima River, which enters just below Richland, tends to lower the concentration along the west shore for a few miles. Opposite the mouth of the Yakima, water is pumped from the Columbia during the summer months in order to irrigate up to about 5,000 acres.

Some 35 miles below the last reactor the Columbia River water is used for municipal supplies by the cities of Pasco and Kennewick. The water systems serve approximately 25,000 people in the two cities. The Snake River enters the Columbia just below Pasco and tends to lower the concentration of the effluent along the east shore for several miles. At McNary Dam, some 30 miles below the mouth of the Snake River, the concentration is essentially uniform.

FIGURE 6

(For figure 7, referred to above, see exhibit 8, p. 191.)

In addition to the water use described above, there are a few industrial plants which pump water from the river. At this time, there is no major navigation on the Columbia River above Pasco, but recreational boating, including water skiing and swimming, is very popular in the Tri-City Area. Private boats are permitted as far upriver as the boundary of the reservation and fishermen have access to the east shore as far upriver as Ringold.

During the fall and winter months the flow in this section of the river amounts to about 80,000 cubic feet per second and is largely controlled by the power dams upstream. In the spring, when the snow melts in the mountains, the flow may increase five fold. During the freshet the current is swifter and thus the travel time between the reactors and the water supplies of Pasco and Kennewick is reduced. While this allows less time for radioactive decay of the effluent isotopes, the exposure is actually reduced because of the greater volume of water available for dilution.

Retention of Effluent Isotopes by Physical And Biological Processes in the Columbia River

The measured amount of radioactive materials in the river water at Pasco is less than the amount calculated from the quantities released from the retention basins. This is because there is accumulation of certain isotopes by living organisms, by suspended silt particles, and by bottom sediments. This depletion may be considered as a type of "self-purification" which has significance in the estimation of the radiation exposure to people who drink the water and in accounting for the accumulation of radioisotopes in aquatic life and river sediments.

After correction for radioactive decay, the average depletion amounts to about 40 per cent between the reactor outfalls and Pasco. The amount of depletion may be different for each particular isotope, however, and values

have been calculated for the 16 individual isotopes which make up 98 per cent of the activity. The values vary from less than 10 to more than 60 per cent. Since this depletion is largely dependent upon the solids content of the river, it is to be expected that the self-purification of the river will vary with stream conditions. Losses of individual isotopes from the water indicate that chromium-51 and zinc-65 should be prominent in down-river sediments and this has been confirmed. No accumulation of long-lived fission products has ever been found in the Columbia River.

Removal of Radioactive Materials by Water Treatment Plants

Very few people, if any, consistently drink untreated water from the Columbia River. Since the river does serve as a source of water for the municipal supplies of Pasco and Kennewick, however, measurements are made of the radioactive materials present before and after the water is processed through the treatment facilities. The city of Pasco uses a flocculation-filtration plant which removes from two-thirds to three-fourths of the gross quantity of radioactive materials present. The Kennewick system is less effective but removes from one-half to two-thirds of the material. Such removal is not considered in calculations used to estimate the degree of exposure associated with the drinking of Columbia River water.

Monitoring of the Radioactive Wastes Released to the Columbia River

Continuous Monitors

The first indications of the concentration of radioactive materials in the effluent are obtained close to the reactor. Samples of the effluent originating from various parts of each reactor are piped to instruments, shown in Figure 8, which scan the samples in sequence for gamma-ray emission, and any off-standard condition is immediately announced to the operating

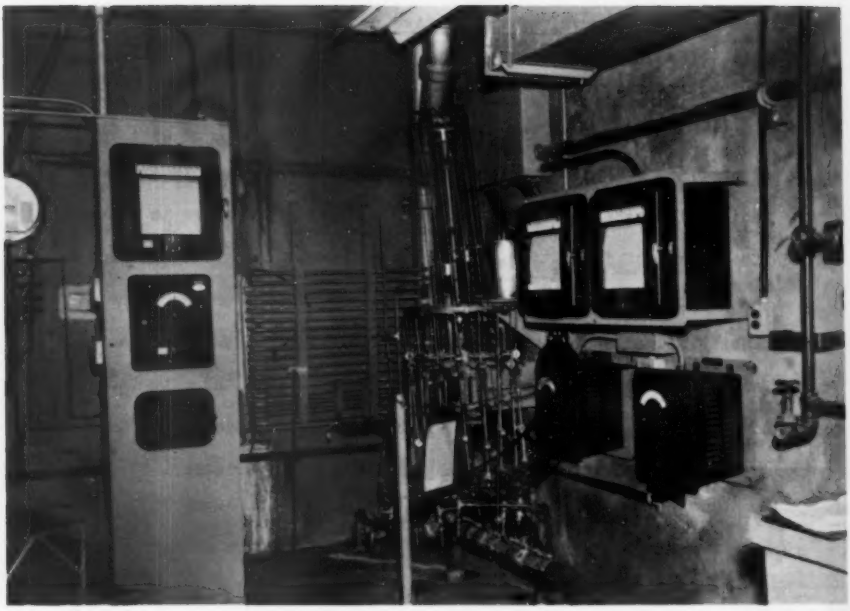

FIGURE 8

Instrument Used to Scan Effluent for Abnormal Radiation

personnel. While the exact source of the radiation is not immediately apparent, any serious trouble, such as the rupture of a fuel element, is quickly noted and can be identified with a particular part of the reactor. The signals from such monitors usually provide the basis for shutting down the reactor in order to discharge the offending element.

Continuous monitoring of the effluent is next obtained at the time it leaves the retention basins. This is accomplished with ionization chambers like the one shown in Figure 9, which are suspended over the basins and which continuously measure the gamma dose rate. The activity which is measured originates mostly from radioisotopes with very short half lives and therefore cannot be used to predict the downstream conditions that may occur some hours later. Nevertheless, surges in activity measured by these chambers are an indication of abnormal conditions and action is taken to explain and, if need be, to correct the cause of such surges.

Effluent Analyses

While the instruments described above are valuable as an early warning system, they do not provide enough information for evaluation of the environmental exposure. This evaluation must be accomplished by more detailed analysis of samples of the water - a control measure which has been steadily improved through the years. When the reactors first began operation, effluent from the basins and water from several points along the river were analyzed for gross beta and alpha emitting isotopes on a daily to weekly basis. As laboratory techniques improved, the frequency of sampling, the number of locations sampled, and the types of analyses performed were increased. By 1953, it was possible to measure the individual concentrations of the most significant isotopes in the effluent with sufficient accuracy to permit evaluation of the contribution which each made to the

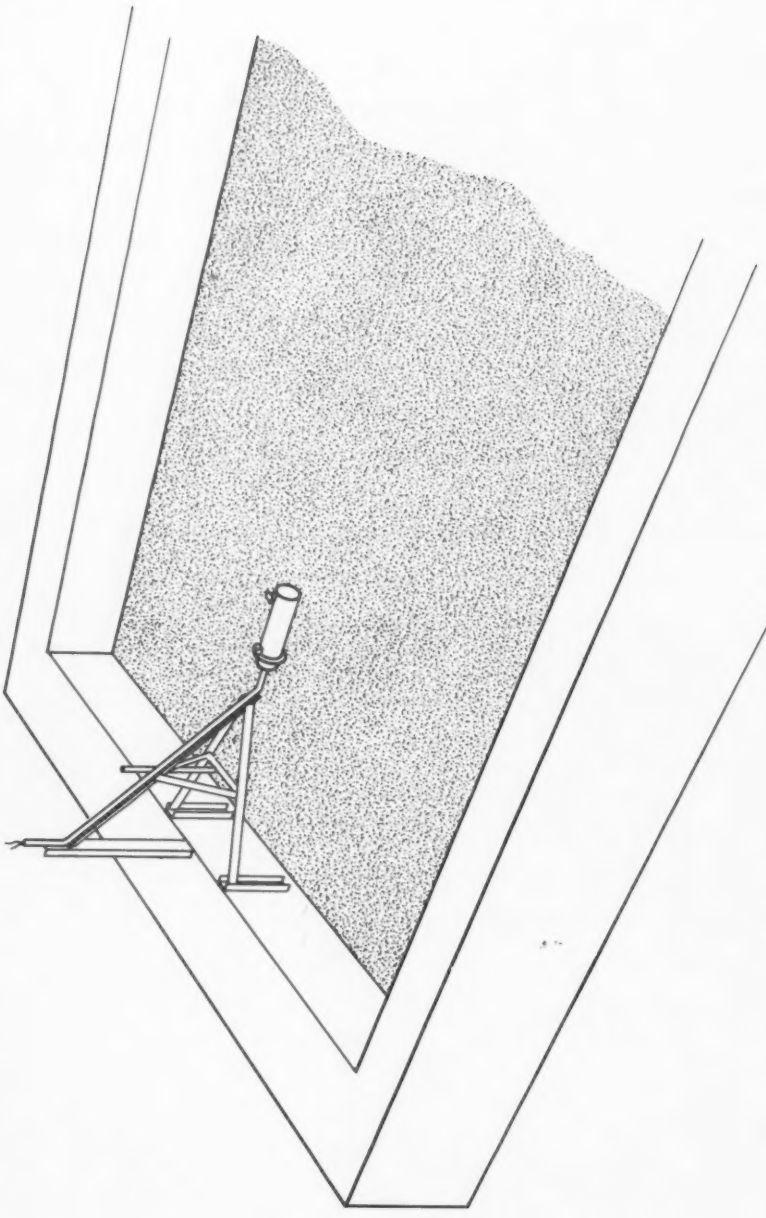

FIGURE 9
INSTRUMENT USED TO MONITOR GAMMA-RAY ACTIVITY OF REACTOR EFFLUENT IN RETENTION BASINS

total activity. The results were extrapolated to Pasco for estimation of the exposure which resulted from drinking Columbia River water. By 1957, a gamma-ray spectrometric system of analysis had been developed to replace many of the tedious and time-consuming chemical separation procedures. This new analytical method also provided the increased sensitivity necessary for convenient analysis of individual radioisotopes in river and sanitary water from Pasco and Kennewick. The present program is shifting to this method since it avoids the uncertainties of extrapolation of results from the several points of discharge.

At the present time, radioisotopic analysis of about 20 isotopes or groups of isotopes (sufficient to define the exposure accurately) are made at weekly intervals on samples of water collected from the Columbia River immediately below the reactors and at Pasco, and from the sanitary water supplies of Pasco, Kennewick, and the most donwriver reactor. These weekly analyses are augmented by analyses on samples obtained from automatic equipment, shown in Figure 10, which collects and combines the water over the entire week's period. The composite samples are especially useful in determining the average concentration of the longer-lived isotopes.

The contribution which each reactor makes to the total isotope load in the river is determined from samples of the effluent which are analyzed each day for gross beta and alpha emitters, each week for uranium, plutonium and polonium, and twice each month for the full 20 isotopes or groups which are of interest to the human exposure limit. Special analyses are also run on samples of the effluent which may contain purge material or rupture debris.

Fish and Waterfowl

The possibility that Columbia River fish might pick up radioactive materials from the effluent was recognized before the first reactors were put into operation during the latter part of 1944, and samples of fish

INSTRUMENT USED TO MONITOR GAMMA-RAY ACTIVITY OF REACTOR EFFLUENT IN RETENTION BASINS

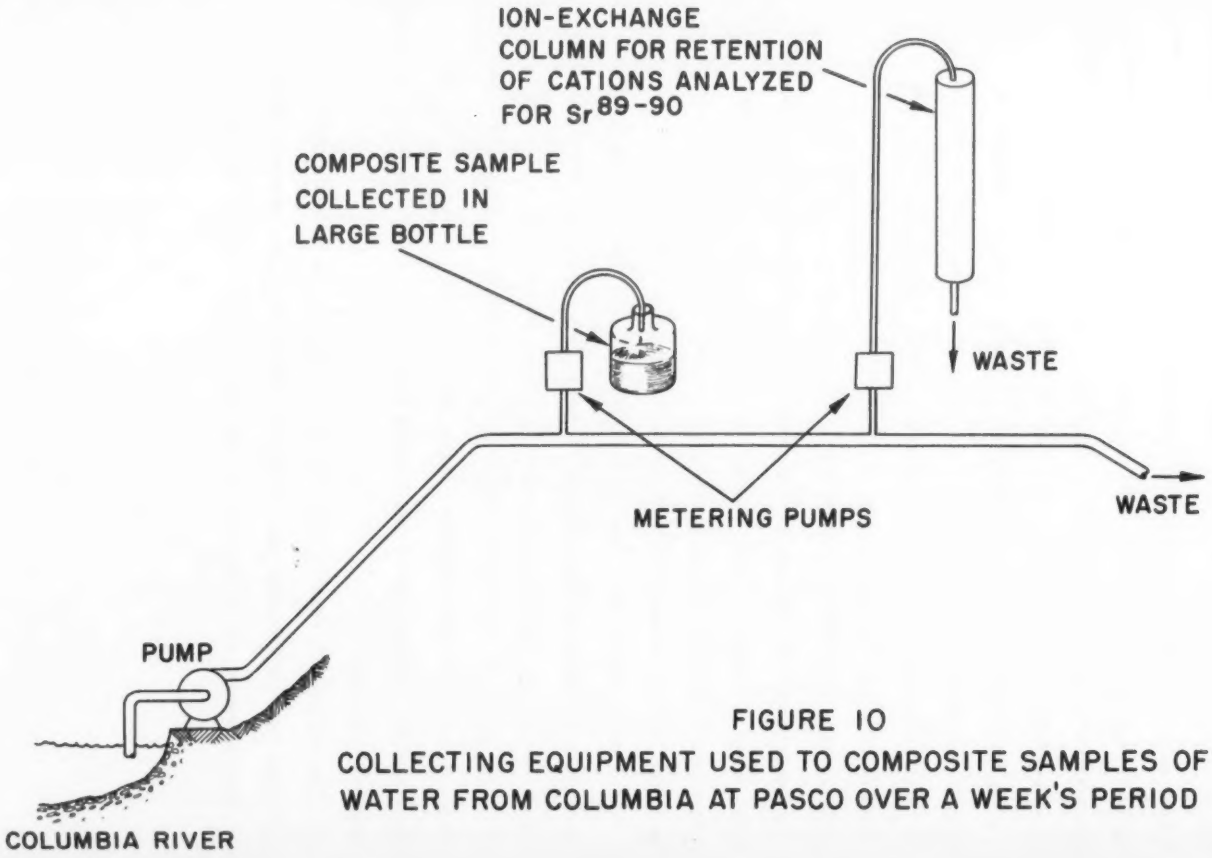

FIGURE 10

COLLECTING EQUIPMENT USED TO COMPOSITE SAMPLES OF WATER FROM COLUMBIA AT PASCO OVER A WEEK'S PERIOD

collected in 1945 showed that this was indeed the case. The characteristics of the radioactivity found in the fish were like those of radiophosphorus (P^{32}) - a result that was not anticipated at the time from information then available on the isotopic composition of the effluent water. A field survey program was started in 1946 to determine which organisms were most significantly involved, what isotopes were accumulated and their sites of deposition, what concentrations occurred in various parts of the river, and how the concentrations changed with time.

The field studies were continued for several years and provided a sufficient amount of data to give reasonably accurate answers to the basic problems. The diatoms suspended in the Columbia River water and the algae which grew on the rocks and other submerged objects usually showed the greatest variety and largest amounts of isotopes which they picked up directly from the water, but the relative abundance of the various isotopes was considerably different from that in the water. Fewer isotopes were found in the small aquatic animals which fed on the algae and other green plants. The fish were found to contain radiophosphorus and a few isotopes of lesser importance, such as radiozinc, which they gradually accumulated from the food organisms which they ate. Although the fish which lived close to the effluent outfalls absorbed a limited amount of radiosodium and some other isotopes directly from the water, this source was found to be much less important to the radioactive contamination of the fish than the food chain route. Figure 11 shows the marked seasonal variation that occurs in the fish and which results not so much from fluctuations in the concentration of effluent in the river as from changes in the metabolic rate of the fish. Some other types of results will be mentioned in connection with research and development work associated with disposal of low level wastes to surface waterways.

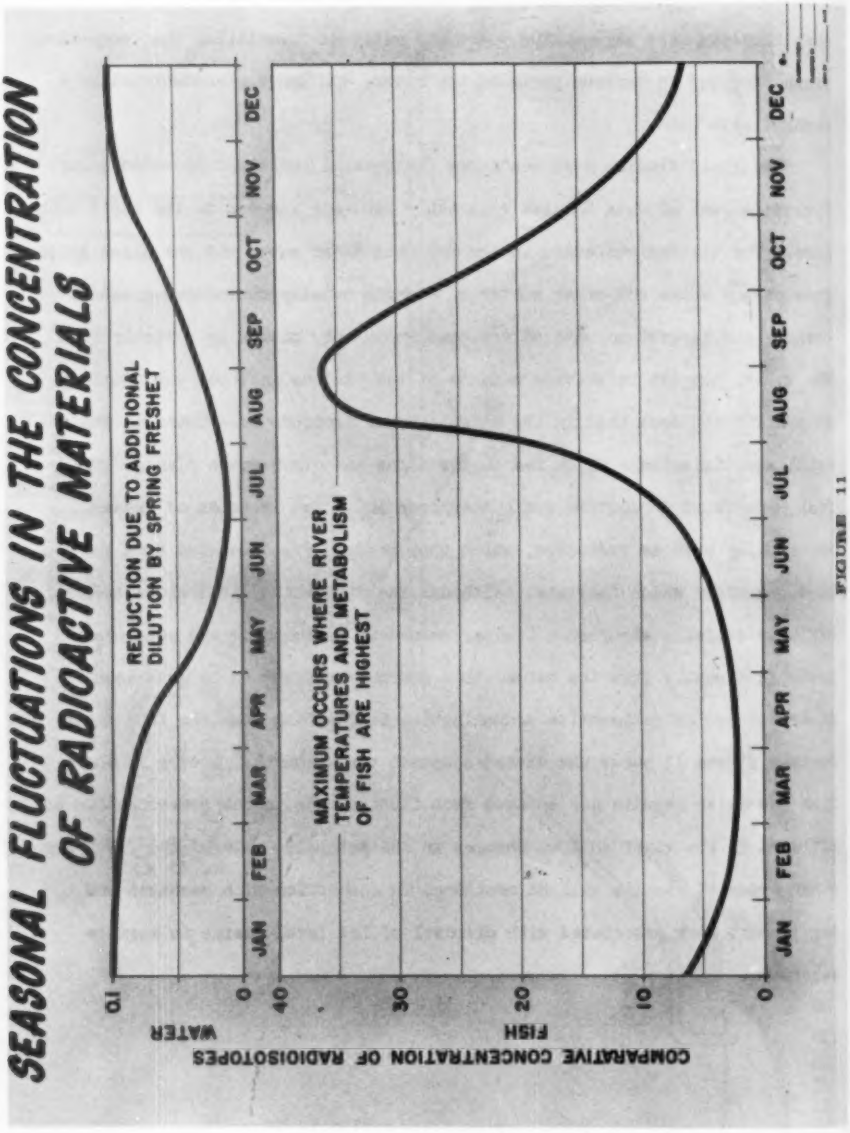
SEASONAL FLUCTUATIONS IN THE CONCENTRATION OF RADIOACTIVE MATERIALS
COMPARATIVE CONCENTRATION OF RADIOISOTOPES
WATER
FISH
0.1
0
40
30
20
10
0
REDUCTION DUE TO ADDITIONAL DILUTION BY SPRING FRESHET
MAXIMUM OCCURS WHERE RIVER TEMPERATURES AND METABOLISM OF FISH ARE HIGHEST
JAN FEB MAR APR MAY JUN JUL AUG SEP OCT NOV DEC

FIGURE 11

From the comprehensive biological surveys of the Columbia River carried out during the early years it was possible to select a few key organisms and sampling sites for continued routine assay of the concentration of effluent isotopes in the aquatic forms. Such measurements are now carried out as a part of the regular monitoring program which is used to evaluate the radiation exposure associated with the operation of the Hanford plants.

Of the several species of fish caught by sportsmen above and below the project boundaries, the Rocky Mountain Whitefish is considered as the most important in relation to radiation exposure. The whitefish spend their entire life in the river and their feeding habits are such that they are among the species which accumulate the largest amounts of radiophosphorus. They move up and down the river to some extent so that contaminated individuals are available both above and below the reservation. Further, these fish are easily caught during the fall months when their concentration of deposited isotopes is greatest. Whitefish and some other species are sampled during the fishing season, and at other times of the year the radioactive content of fish is followed from minnows which are readily available throughout the year.

The salmon of the Columbia River are not significantly involved in the uptake of radioactive materials because of their migratory habits. The young salmon migrate to the ocean before they have had an opportunity to accumulate a maximum amount of the effluent isotopes and once in the ocean they feed on non-radioactive food and soon lose what little contamination they had picked up. The large salmon caught by fishermen have only background amounts of radiation. When the adult salmon re-enter the Columbia on their spawning migration, they consume virtually no food and thus no radioactive materials.

Since the Columbia River supports fish which are of considerable commercial and sports value, steps are also taken to assure that the aquatic forms are not affected by the effluent. Much of this work is of a research nature, but one phase has been established as a unique type of effluent monitoring. A continuous sample of the effluent being discharged into the river is diverted into an aquatic biology laboratory. The effluent is diluted to various concentrations with river water in troughs which contain small salmon or other local species of fish. The equipment is shown in Figure 12. Continuous exposure of the fish for extended periods of time has clearly demonstrated that the concentration of effluent in the Columbia River is well below the toxic level. Since this monitoring with fish has been virtually uninterrupted since the reactors first began production, we also have the assurance that there have been no operational irregularities or accidents which might have resulted in fishkills in the Columbia River.

Waterfowl which visit the Hanford Reservation also pick up radioactive materials from food organisms which they obtain from the river. A comparatively few ducks remain on the river throughout the year so that the accumulation of radiophosphorus and a few other isotopes builds up to a level which is in equilibrium with their environment. These ducks are not available to off-project hunters until the fall hunting season at which time their number is diluted 200 to 1000 fold by flocks of ducks which are migrating through the region. The migratory birds obtain a part of their food from the river but few remain in the area long enough for radioisotopes to build up in the tissues to the greatest concentration possible. Samples of the duck population are taken for radioassay during the hunting season and at other times of the year when the concentrations of isotopes are known to be greatest.

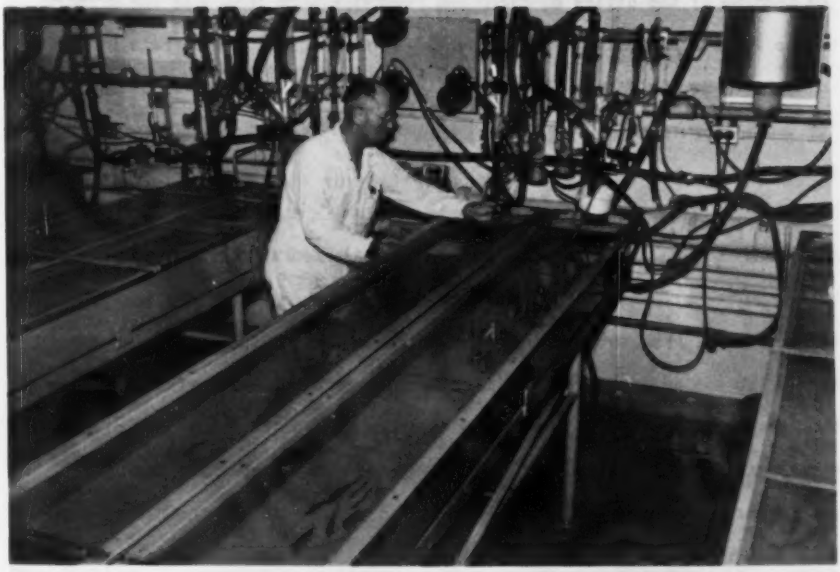

FIGURE 12

Troughs with Young Salmon Used to
Continuously Monitor Reactor Effluent

Radiation Exposure

Evaluation and control of the exposure associated with radioactive wastes from the Hanford plants is based on a philosophy whereby the concentration of particular radioactive materials at the points of exposure, or use, is related to appropriate limits. In the following paragraphs the maximum permissible limits used are one-tenth of the values published by the National Committee on Radiation Protection and Measurement for occupational exposure. The water supply intakes of the cities of Pasco and Kennewick were chosen as the primary points for evaluation of the concentration of radioactive materials in the Columbia River since these towns are the first to make major use of the water for domestic purposes and exposures at locations further downriver will be lower.

Careful evaluation of the effluent problem in the Columbia River requires that each individual isotope be separately appraised in terms of its contribution to the exposure. The sum of the individual contributions must then be taken to obtain the total radiation exposure of the mixture. As already stated, it was sufficient at one time to analyze samples of effluent collected from the retention basins and extrapolate the results to the points of use. While analyses of the effluent water from each of the retention basins are still made in order to appraise the output of each reactor, the recent improvements in analytical methods now provide a practical means for routine analysis of river water from the points of use and eliminate uncertainties associated with extrapolation. Figure 13 shows the relative importance of the various isotopes or groups of isotopes which contribute significantly to the gastrointestinal exposure of persons who drink Columbia River water. The two diagrams show how the relative importance of each individual isotope changes with time. Further changes in relative importance are created by "clean up" in the river as the material moves downstream.

RELATIVE IMPORTANCE OF ISOTOPES
TO EXPOSURE OF GI TRACT
24 HOURS AFTER LEAVING
A REACTOR
(REDUCTION IN TOTAL
EXPOSURE IS IN PRO-
PORTION TO AREAS OF
THE DIAGRAMS.)
4 HOURS AFTER
LEAVING REACTOR
Cu64
RARE EARTHS
Np239
Mn56
OTHERS
p32
Na24
Ga72
Zn69
As76

FIGURE 13

A comparison of Figures 13 and 4 will further bring out the important difference between the relative abundance of each isotope and its contribution to human exposure. For the year 1958, it is estimated that the exposure to the gastrointestinal tract of persons who consistently drank Columbia River water was not over 20 per cent of the maximum permissible limit. The dosage rates actually received from drinking water by virtually all the people are 25 to 50 per cent lower than this owing to the influence of the water treatment plants which remove substantial amounts of the significant isotopes present.

Drinking water represents one, but in some cases, not the only source of exposure which results from the reactor effluent. Some of the research and development work which has gone into determining and evaluating these other exposure sources will be covered in a later topic but their importance warrants that they also be mentioned here. The concentration of radiophosphorus by fish and waterfowl constitutes a source of exposure for those individuals who fish and hunt in the vicinity of the plant. The whitefish is the species currently considered to be of greatest significance because of its level of radioactivity and because it is easily caught from October through January. Our best estimate of the average phosphorus-32 burden in the bodies of individuals who eat whitefish caught at the closest public approach downriver from the reactors amounts to about 10 per cent of the maximum permissible limit. For fish caught at the closest public approach upriver from the reactors, the estimate amounts to about one per cent of the maximum permissible limit. It is conceivable that a very few individuals eat enough fish so that their average body burden of phosphorus-32 rises above 10 per cent of the M. P. L., but highly unlikely that any one person would routinely eat so great an amount that his body burden would approach the M.P.L.

The phosphorus-32 content of the flesh of ducks which are killed along the river within the Hanford reservation is similar to that found in the fish. The number of resident ducks which are harvested by sportsmen is considerably less than the number of whitefish, and ducks may be considered as an alternate source of phosphorus-32 for the purpose of exposure evaluation.

Another potential source of exposure from the isotopes contained in the reactor effluent is the crops which are irrigated with Columbia River water. Since the accumulation of isotopes in these crops is small, the exposure is insignificant. The exposure to persons who eat such foodstuffs is difficult to estimate because of the variability of the diet between individuals and the widespread distribution and seasonal nature of the foods. It is estimated, however, that the human exposure from reactor effluent isotopes which deposit in crops is less than a tenth of one per cent of the permissible limit.

Besides the internal exposure which is received by people who drink Columbia River water or eat foods which contain radioactive materials, some people receive a small external exposure dose while boating on the river or while swimming. This is estimated from small ionization chambers placed in the river. The additional exposure to the limited number of people who participate enthusiastically in such sports is low since the exposure amounts to only about five per cent of the off-plant maximum permissible limit.

From the extensive data available, it appears that the phosphorus-32 content of fish and waterfowl can contribute to individuals the highest single source of exposure which results from the release of reactor effluent into the river. The numbers of people who eat such fish and ducks will be relatively small and the exposure will be quite seasonal.

PART 3

RELEASE OF GASES, VAPORS, AND PARTICLES TO THE ATMOSPHERE

R. E. Tomlinson

The controlled release of gaseous radioactive wastes to the atmosphere at the Hanford Plant has been based on a policy of managing waste disposal as an integral part of the production process. Effective management of atmospheric disposal of radioactive wastes has required activities in at least four different areas:

1. Operation and maintenance of existing process systems and controls such that the release of radioactive materials is held within established limits;
2. Research and development of improved processes, equipment, and instrumentation, as required to meet radiation protection criteria;
3. Radiation monitoring and training to protect the project workers, and regional monitoring to assure that internal and external exposures of people, plants, and animals in the project environs are below appropriate nationally recommended maximum permissible limits or self-imposed Hanford control limits;
4. Basic and applied research on the biological effects of ionizing radiations, modes of exposure, and the determination or refinement of maximum permissible limits and protection criteria.

The Hanford Area

As mentioned in Part 1, the Hanford project is in a saucer-like basin, surrounded by hills and mountains up to 3600 feet above sea level. The chemical separations plants, in which plutonium and uranium are separated from the fission products, are located on a plateau in the center of the project

about 15 miles from residential areas and farm lands. Temperatures range from -25° F to 110° F, and rainfall averages about 7-1/2 inches per year. The winds vary from calm to 30 to 40 miles per hour frequently, and generally attain 50 to 60 miles per hour at the 50-foot level at least once a year. The region is semi-desert, with a natural ground cover of grass and sagebrush which is used for sheep and cattle range outside of the project. Surrounding farmlands are irrigated for pasture and produce at the lower elevations, and are used for dry-land grain growing at higher elevations.

Figure 1 shows prevailing wind directions in the Hanford environs. Several major gaps through the surrounding mountains are important because of the natural channeling of air flow through them. The Beverly Gap at the northwest corner is a narrow gorge cut through the Saddle Mountains by the Columbia River. The Ringold Gap on the east side permits drainage of the extensive rising ground to the east. The broad river valley leading southeastward past Richland permits generally unobstructed flow of air from the project. Any airborne radioactive material from the separations area is generally carried in an easterly direction.

A meteorology station with a 400-foot tower is continuously manned in the separations areas, and is connected by teletype with the nationwide weather network. Supplemental data such as wind velocity and direction, and temperatures, are provided from the reactor areas, fuel preparation area, and city of Richland. Unmanned stations are scattered elsewhere throughout the project, and are soon to be converted to wind-powered radio-telemetering for routine and emergency interrogation by the master station at the meteorology tower. At that time, radiation instruments will also be included in the remote stations.

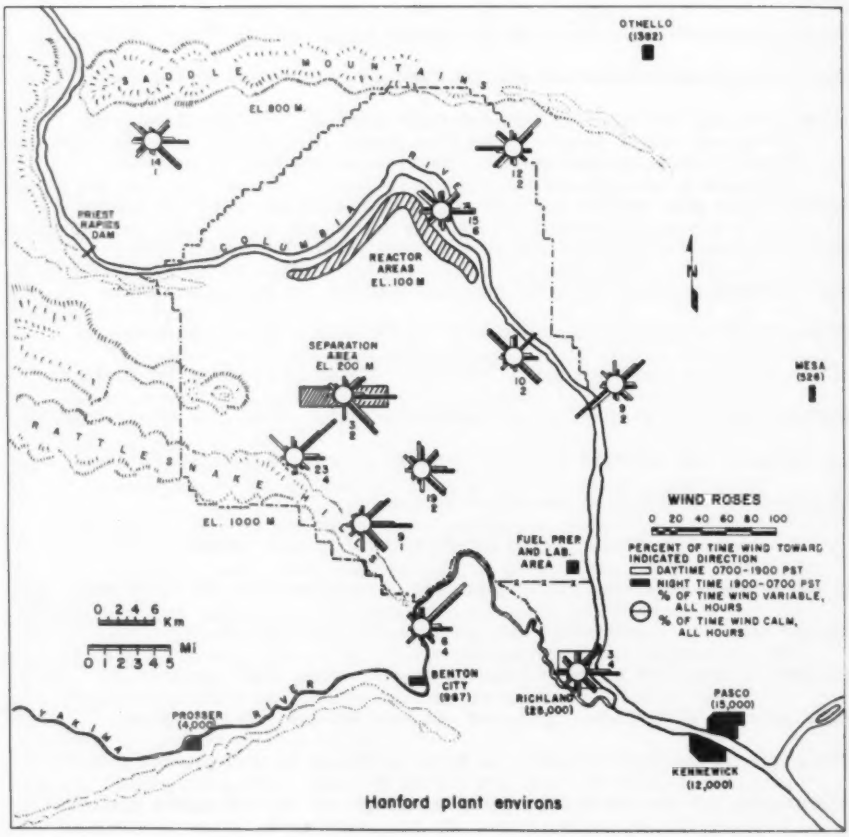
OTHELLO
(1382)
SADDLE MOUNTAINS
EL. 800 M.
PRIEST RAPIDS DAM
COLUMBIA RIVER
REACTOR AREAS
EL. 100 M
SEPARATION AREA
EL. 200 M
MESA
(526)
RATTLESNAKE HILLS
EL. 1000 M
FUEL PREP AND LAB. AREA
WIND ROSES
0 20 40 60 80 100
PERCENT OF TIME WIND TOWARD INDICATED DIRECTION
DAYTIME 0700-1900 PST
NIGHT TIME 1900-0700 PST
% OF TIME WIND VARIABLE, ALL HOURS
% OF TIME WIND CALM, ALL HOURS
0 2 4 6 Km
0 1 2 3 4 5 MI
BENTON CITY
(987)
RICHLAND
(28,000)
PASCO
(15,000)
KENNEWICK
(12,000)
PROSSER
(4,000)
YAKIMA RIVER
Hanford plant environs

FIGURE 1.

Operational Experience and Improvement

The gaseous effluents from the separations plants are exhausted to the atmosphere through 200-foot concrete stacks having steel liners. Shortly after startup of the two original bismuth phosphate plants in 1945, radioactive iodine-131 was identified on samples of vegetation collected within the project around the stacks. The radioiodines are fission products formed during irradiation of the uranium fuel elements, and partially released when the elements are dissolved in nitric acid. Iodine-131 has a radioactive half-life of eight days.

To prevent unnecessary exposure to life in the environs, the cooling period between reactor discharge and fuel element dissolution was increased, even though this action temporarily delayed plutonium production. In addition, controls were established such that dissolving could be performed only during weather conditions favorable for iodine dispersion. In a short time, equipment was designed and installed to remove more than 99.5 per cent of the radioiodine from the dissolver off-gas before it was vented to the atmosphere. This equipment, shown schematically in Figure 2, consists of a gas heater to raise the temperature to about 200° C, and a column containing a porous packing material impregnated with silver nitrate. The heated iodine chemically combines with the silver and is retained in the column. When nearly exhausted, the column is regenerated by adding more silver nitrate and the liquid waste, which contains the iodine-131, is sent to the underground waste storage tanks. When this equipment was operating satisfactorily, cooling times for the fuel elements were again shortened and the meteorological controls were relaxed. All dissolving equipment installed subsequently has been equipped with silver reactors. The capital cost of a heater-reactor unit runs about twenty to

(For figure 2, referred to above, see exhibit 9, p. 192.)

fifty thousand dollars. In 1957, absorbers were installed to recover nitric acid from the dissolver off-gases. These units also scrub iodine from the effluent, the iodine being returned to process with the recovered acid.

During the past decade extensive experiments were performed involving the daily feeding of varying amounts of radioiodine (I^{131}) to sheep, a representative grazing animal of this region. Feeding extended throughout several generations at levels (ranging from 0.15 to 5 /uc per day) which were selected so as to lend a firm basis for the establishment of a maximum permissible concentration for vegetation. No damage was observed following the feeding of 0.15 /uc per day which gives a dose rate of about 3 rad per week to the thyroid, the gland which contains the highest concentration of radioiodine. To date the only effect noted following the feeding of 0.5 and 1.5 /uc per day is a depression of thyroid uptake of I^{131}. Thyroid tumors were observed in the majority of the animals fed 5 /uc per day for five to seven years which followed exposure to I^{131} during fetal life and during lactation by way of their mothers, which were also fed 5 /uc per day. Since no damage was observed at the 0.15 /uc per day level, this daily intake was related to the consumption of pasturage by sheep and a limit of 10^{-5} /uc/gm of vegetation was derived. This limit is also believed to be applicable to garden produce eaten by man and for the ingestion of milk produced by cows which graze on such pasturage. From the limit on vegetation, a maximum permissible concentration (MPC) in air can be calculated, based on empirical measurements of deposition factors. The recommended MPC derived at Hanford in this fashion, is 10^{-13} /uc/cc. By comparison, the nationally recommended MPC for air breathed by humans in non-controlled areas is 6×10^{-10} /uc/cc.

The results of continual sampling and analysis of exhaust air, the environmental air, and vegetation showed that the vegetation limit could be met, on the average, if the total emission of iodine-131 was not allowed to exceed ten

curies per week. This release rate was accepted as an operating standard which is still in effect. The success of the process controls is illustrated in Table 1.

Table 1

Atmospheric Disposal of Radioactive Iodine
From the Separations Plants Stacks

Year	Curies iodine-131/day
1953	2.0
1954	1.5
1955	3.2
1956	1.0
1957	1.0
1958	1.2

Figure 3 shows the I^{131} deposition pattern, averaged over a period of several months and normalized to an emission rate of one curie per day. This illustrates the adequacy of the control of I^{131} in the environs.

The necessity of installing filters for the large-volume ventilation exhaust air was first recognized in 1947, when radioactive particles were found on the ground around the stacks. The original exhaust air blowers between the plant and stack were made of mild steel, painted with an asphalt-base paint. After a time, rust spreading beneath the paint caused flakes up to several inches across to be torn off and blown out of the stack. The contamination was local and minimal, though as a precaution operating personnel were protected against inhalation of the particles until the potential hazard was eliminated. Large filter beds of graded layers of sand were constructed, through which the building exhaust air was routed before release via the stack. In the more recently constructed Purex separation plant the filter bed is made of graded mats of glass-fiber. A sand filter, designed to handle about 40,000 cubic feet of air per minute, costs about $575,000. The glass-fiber filter handles 120,000 cfm and costs about $600,000. Either type of

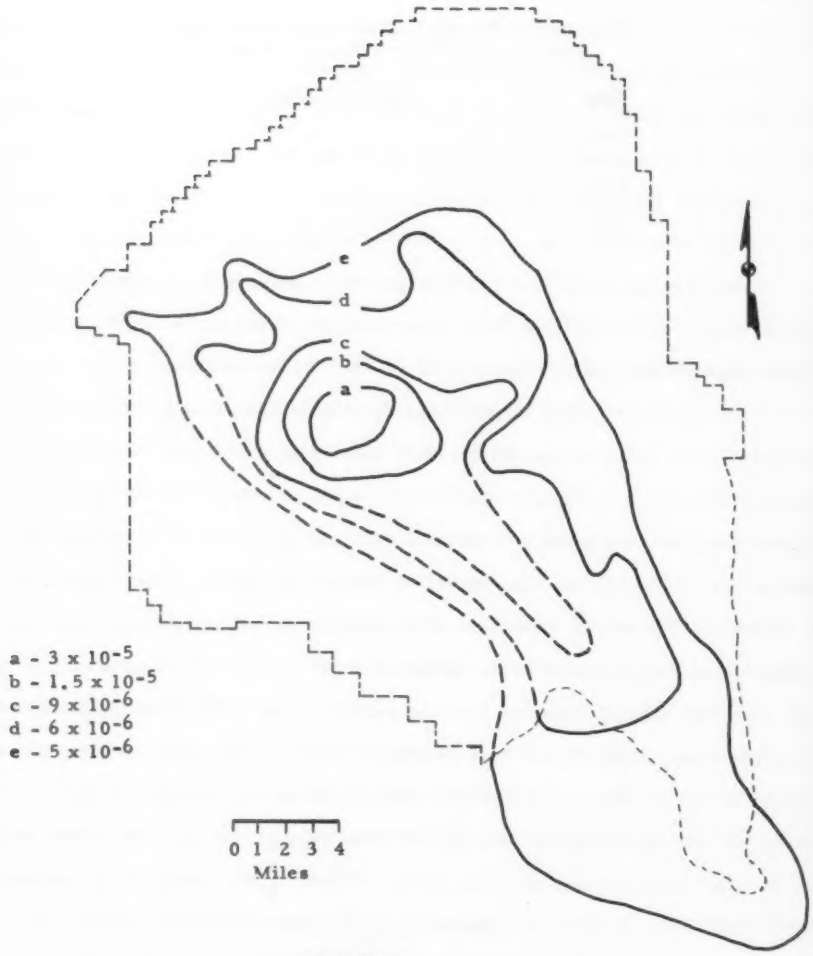

FIGURE 3
Average I^{131} deposition - μc/gm of vegetation per curie/day emitted

filter retains more than 99.5 per cent of the particulate radioactive materials entering the unit, with the glass-fiber unit being somewhat more efficient. Either type performs satisfactorily so long as the humidity of the ventilation air is kept well below saturation.

Another radioactive fission product which required specific study is ruthenium. Beginning in 1952, ruthenium-103 and 106 were found to be escaping to the environs, and were detected on the ground and vegetation around the stacks as well as in the stack gas samples. In the Redox process, the bulk of the ruthenium was removed from the product solution by volatilization as ruthenium tetraoxide. While most of the ruthenium was absorbed in caustic scrubbers, some decomposed to form ruthenium dioxide, which is essentially chemically inert. Minute quantities of the ruthenium were thus escaping the caustic scrubber and exhaust air filter and were being vented to the atmosphere as tiny particles.

Improvements were made in the process and in the ruthenium removal equipment. In addition, this off-gas was routed with the ventilation air exhaust through the sand filter. The sand filter not only removes ruthenium particles, but also acts as an absorber for any ruthenium tetraoxide vapor remaining in the air. Since these changes, ruthenium emissions have been insignificant.

Smaller emissions of gross fission products continued, but were associated with flakes of ammonium nitrate, up to several inches across. These flakes were being formed on the stack liner by the chemical reaction of ammonia, formed during the dissolution of the aluminum jackets with caustic, combining with the nitrogen oxides evolved during the subsequent uranium dissolution step. The effective countermeasure has been to wash down the stack liner about once a week with water from spray rings installed at the top of the liner in mid-1954.

The plutonium purification facilities were originally equipped with process off-gas filters, using the paper filters developed by the Chemical Warfare Service. In 1957, all such filters in the expanded purification facilities were replaced with fire-resistant units developed especially for this purpose. No significant contamination of the atmosphere has ever been experienced from this plant.

Regional Monitoring

At the present time only minute amounts of mixed fission products escape with the gaseous effluents. The stack gas is sampled by a moving asbestos filter strip, which is continually scanned by a recording scintillation-type monitor for beta emitters, alpha emitters, and specifically ruthenium-103 and 106; the gas is also sampled for iodine-131 by a continuous countercurrent-flow caustic scrubber, and the iodine-131 concentration is continuously recorded. Filter samples which show a high beta measurement are removed for detailed isotopic analyses. Samples from the Redox and Purex stacks showed an average total daily emission in 1957 of 1.0 curie iodine-131, 0.02 curie rare earths plus yttrium, 0.01 curie niobium-95, 0.004 curie zirconium-95, 0.004 curie ruthenium-103, 0.002 curie ruthenium-106, 0.004 curie strontium-89, and 0.0005 curie strontium-90. Beta emitters other than iodine-131 averaged 0.044 curies per day. These results are typical, and vary little from year to year. At the plutonium purification plant, continuous sample filters are replaced and measured periodically for gross alpha, while at the uranium oxide plant similar samples are changed semi-weekly and measured for uranium. The amount of plutonium and uranium released to the atmosphere is insignificant.

In addition to monitoring the gaseous wastes at the point of discharge, representative locations in the environs are periodically surveyed and sampled to determine the pattern and intensity of environmental contamination. This

program involves some 10,000 samples taken from gaseous or atmospheric sources each year. The particulate aerosols are measured by filter paper samples taken from 31 on-plant locations and 13 off-plant locations. The iodine-131 is determined at 14 locations by means of caustic scrubbers through which the air is pulled. The perimeter points are mostly concentrated in the Richland, Kennewick, Pasco area, while those at greater than 100 miles are primarily in southern Oregon, southern Idaho, and western Montana. Typical results of this type of analysis are given in Table 2:

Table 2

Average Measured Air Concentrations - 1958

Month	I^{131} Perimeter	Alpha Emitters Perimeter	Beta Emitters Perimeter	Beta Emitters 40 - 100 mi.	Beta Emitters >100 mi.	Particulates Perimeter	Particulates 40 - 100 mi.	Particulates >100 mi.
	units of 10^{-13} microcurie per cc					particles per cubic meter		
January	0.8	<0.02	4.7	4.5	4.9	0.03	0.03	0.04
February	1.0	<0.02	3.6	2.6	2.7	0.01	0.02	0.03
March	0.6	<0.02	6.0	5.5	6.6	0.006	0.02	0.01
April	2.8	<0.02	30	22	22	0.04	0.08	0.10
May	2.0	0.02	18	13	14	0.05	0.08	0.10
June	3.0	0.02	17	9	9	0.03	0.09	0.07
July	2.1	0.02	14	10	10	0.05	0.08	0.07
August	2.0	<0.02	8.6	13	9	0.09	0.07	0.08
September	1.1	0.02	7.2	6.3	6.3	0.02	0.04	0.03
October	3.1	0.03	8.2	13	25	0.03	0.05	0.06
November	8.5	0.05	21	32	42	0.06	0.16	0.22
December	1.2	0.03	14	8	16	0.05	0.08	0.12
AVERAGE	2.4	0.02	12	12	14	0.04	0.07	0.08

The quantities of radioactive material deposited on the ground and the resulting radiation exposures are estimated from measurements taken with ionization chambers located three to seven feet above the ground, by obtaining over 11,000 samples of vegetation per year at distances up to 250 miles from the plant, and by surveying with portable instruments specified areas both in the vicinity of the plant and in the environs. Typical results of the vegetation analyses are given in Table 3.

Table 3

Vegetation Contamination - 1958
Units of 10^{-6}/uc/gm wet weight

	On-Plant Site			Perimeter			40 - 150 miles	
Month	I^{131}	Beta Emitters	Alpha Emitters	I^{131}	Beta Emitters	Alpha Emitters	I^{131}	Beta Emitters
January	3.3	110	3.1	2.3	92	0.1	<1.5	88
February	5.5	94	0.8	1.9	71	0.09	<1.5	72
March	6.7	81	0.7	<1.5	42	0.07	<1.5	85
April	14.	290	0.3	6.8	150	0.04	19.	350
May	2.7	130	0.1	<1.5	61	<0.03	<1.5	77
June	2.6	100	0.2	<1.5	71	<0.03	<1.5	44
July	4.0	100	0.1	2.6	76	0.05	2.1	86
August	3.2	110	<0.1	<1.5	56	0.04	<1.5	62
September	1.6	71	0.1	<1.5	38	0.04	<1.5	28
October	2.6	120	0.2	<1.5	67	0.08	3.3	170
November	8.2	270	0.4	5.6	200	0.1	10.0	270
December	5.4	230	0.6	<1.5	210	0.14	<1.5	200

Summary of Hanford Experience

The original wartime construction of the Hanford separations plants relied on dispersion distance to protect the surrounding environs from radioactive materials escaping from the processes to the atmosphere. The necessity of containing plutonium in the purification plant was recognized and filters were installed in the ventilation exhaust when the plant was built. When conditions arose which had potential for significant off-site exposure from the bismuth phosphate plants, corrective measures were applied. These measures were incorporated in the design of the replacement solvent extraction plants, but the new processes and increasing production rates engendered new atmospheric release problems. Intensive programs of basic and applied research and development in the biological, chemical, physical, and radiological sciences, were combined with sound engineering and careful operation, to overcome the problems as they arose. Efforts are continuing on all fronts to

define more accurately the essential parameters and criteria in order to improve still further the economies in plutonium production and to minimize environmental and occupational exposures.

Radiation exposures contributed by the release of radioactive material from the Hanford plants to the atmosphere have been well within the maximum permissible limits recommended by national authorities for people in the vicinity of such plants. A summary of environmental exposure, including all major modes of exposure to the most critical organs of humans was included in Part 1.

PART 4

STORAGE OF HIGH-LEVEL FISSION PRODUCT WASTES

R. E. Tomlinson

In the separations processes utilized at Hanford, nearly all of the fission products formed by nuclear fission are segregated and stored as a high-level chemical waste for which there is no immediate use. The technology evolved in the storage of these wastes is discussed in this paper, with particular emphasis on the safety aspects of this operation.

The Nature of the Wastes

From the beginning of plant operation to the present time, four primary separation processes have been employed at Hanford. These processes include the bismuth phosphate ($BiPO_4$) precipitation process for plutonium recovery, the TBP (tri butyl phosphate) solvent extraction process for the recovery of uranium from the $BiPO_4$ "metal wastes", and the Redox and Purex solvent extraction processes for the recovery of uranium and plutonium. Because of security restrictions, hypothetical data have been substituted for actual data whenever plutonium production could be estimated from the information presented. Hanford is now preparing to reprocess the irradiated fuels from some of the industrial power reactors, using the Redox process for the decontamination and recovery of the fissionable materials.

Chemical and Physical Characteristics

The high-level wastes from the reprocessing of nuclear fuels are stored as alkaline slurries, the volumes and compositions of which

are dependent upon the specific processing conditions in use at the time. Typical waste compositions arising from past and current plant operations are presented in Table 1.

The quantity of precipitated salts in the stored waste is variable, depending on the separations process and the temperature of storage. In the $BiPO_4$ metal wastes, the uranium slowly formed a bulky precipitate, the solubility of which was decreased by an increase of temperature. In the Redox and Purex wastes, uranium forms a dense precipitate which occupies little volume and is probably fast-settling. Iron and chromium form flocculent, slow-settling precipitates which may be kept in suspension by agitation. These precipitates carry up to half of the radioactivity by adsorption, and most of the heat of radioactive decay is evolved in their vicinity. While the sodium aluminate in Redox and decladding wastes is very soluble at high temperatures, the salt slowly decomposes into sodium hydroxide and a hydrated aluminum oxide of indefinite composition. Once precipitated, this gelatinous oxide is slowly dehydrated to form a dense, granular material which is insoluble in acid or alkali. This action produces a sludge layer which slowly grows in density and depth with time; the action is largely beyond operator control, although laboratory data indicate that the precipitation proceeds more slowly as the temperature is increased. The depth of the sludge layers observed in the storage tanks ranges from barely detectable in continuously agitated Purex waste systems to several feet in non-agitated tanks of aged Redox waste.

Table 1

Typical Compositions of Wastes from the Reprocessing of Nuclear Fuels

Type Waste	High Level			Decontamination	Decladding
Process	Redox	Purex	$BiPO_4$	$BiPO_4$	Caustic
Composition, molarity					
Na^+,	6.4	2.4	4.1	4.1	4.9
Bi^{+++},			trace	trace	-
Fe^{+++},	trace	trace	-	trace	-
U,	trace	trace	-	trace	-
AlO_2^-,	1.4	-	-	trace	1.2
OH^-,	0.8	0.1	-	0.2	1.0
NO_3^-,	4.2	2.1	0.6	2.5	0.6
NO_2^-,	-	-	-	-	0.9
$SO_4^=$,	trace	trace	0.25	0.1	-
$CO_3^=$,	-	0.1	1.0	-	-
$CrO_3^=$,	trace	trace	-	-	-
SiO_3,	-	-	-	-	trace
$HPO_4^=$,	-	-	0.27	0.5	-
Specific Gravity					
	1.32	1.10	1.25	1.15	1.18
Fission Products, approximate per cent contained					
	100	100	90	10	0.1

The reprocessing of nuclear fuels designed primarily for the production of power will give rise to new types of radioactive wastes. The fissionable materials are to be recovered by the Redox process, and the high-level wastes from declad elements will, therefore, be quite similar to those described above. The cladding materials of the new fuels will differ from those of current fuels, however, both in composition and in quantity per unit of fuel processed. Alloys of aluminum, zirconium, and stainless steel are currently being planned as cladding materials for the power fuels. The weight of the cladding, excluding the massive end fittings, runs as high as one-third the weight of the fissionable material.

Since the plans for the reprocessing of these fuels have not yet been crystalized, the nature of the wastes cannot be described in detail. One processing scheme employs a size reduction (either chopping or sawing) of the fuel elements, followed by the dissolution of the fissionable materials from the pieces without dissolving the cladding material. The cladding material would then be either removed and stored in metallic form, or dissolved and stored in tanks as alkaline slurries. An alternative plan employs the sequential dissolution of the cladding and fissionable materials, with the chemical wastes from the decladding and solvent extraction operations being handled separately. Still another plan employs the simultaneous dissolution of the cladding materials being handled together. The process and engineering feasibility of each alternative plan has been established, and the choice of plan will be based on economic and operational factors currently being studied.

The nature of the solutions obtained by the chemical dissolution of the various cladding materials is presented in Table 2. The information is based on development data, and is presented on the premise that the cladding material is dissolved in the absence of fissionable materials. It would be our intention to treat any such solutions with sodium hydroxide as required to store them in the underground tanks as alkaline slurries. The Niflex and Darex processes, indicated on Table 2, could be used for the simultaneous dissolution of cladding and fissionable materials, so these wastes might be blended with the fission products and the process reagents from the Redox recovery operations.

Radiochemical Composition

The radioisotopes produced by the fission of uranium are such that they must be positively contained for many hundreds of years. Consider, for example, the fission products present in a ton of uranium containing an initial enrichment of 1.2% U-235, after irradiation at 5 megawatts per ton to a total exposure of 2500 megawatt days per ton. The concentrations of significant isotopes are presented in Table 3, on the assumption all of the fission products were segregated in 100 gallons of waste for storage. Each concentration is then compared with the respective maximum permissible concentration (MPC) for that isotope in drinking water. Any unrecovered plutonium and other transuranic elements are also present and must be positively contained with the fission products.

Table 2

Typical Process Solutions for the Decladding of Fuel Elements

Process*	Zirflex	Sulfex	Niflex	Darex	Caustic
Reagents	NH_4NO_3 NH_4F	H_2SO_4	HNO_3 NH_4F	HNO_3 HCl	$NaOH$ $NaNO_3$
Cladding Material	Zirconium	Stainless Steel	Stainless[1] Steel	Stainless Steel	Aluminum
Solution Composition, molarity					
NH_4^+,	2.3	-	1.0	-	Na^+ 4.9
H^+,	-	0.5	0.8	2.5	AlO_2^- 1.2
$SO_4^=$,	-	2.5	-	-	OH^- 1.0
Fe^{+++}, Cr^{+++}, Ni^{++},	-	1.0	0.4	1.5	NO_3^- 0.6
$ZrF_6^=$,	0.6	-	-	-	NO_2^- 0.9
F^-,	0.6	-	2.0	-	
NO_3^-,	0.5	-	0.8	3.7	
Cl^-,	-	-	-	3.2	
Solution Volume, gallons per pound of cladding[2]	2.2	2.2	5.5	1.5	3.5
Specific Gravity,	1.09	1.16	1.09	1.3	1.18

(1) Zirconium and aluminum could also be dissolved by variations of this process, but the zirflex and caustic processes, respectively would probably be preferred.

(2) Solution volume per unit weight of cladding material may be increased if larger volume is required to cover the elements in the dissolver.

*These processes are each given a simple name to facilitate identification and reference to them. These names mean:

Zirflex = Zirconium dissolved in an alkaline solution containing fluoride ion.
Sulfex = Sulfuric acid used as solvent.
Niflex = Nitric acid and hydrofluoric acids used as solvent.
Darex = Dilute Aqua Regia used as solvent.

Table 3

Concentration of Significant Radioisotopes in the Liquid Waste from Irradiated Uranium

Nuclide	Half-Life Years	Concentration(1) /uc/cc	MPC(2) /uc/cc of H_2O	Number of Times Above MPC
Sr-89	0.148	1.6×10^{4}	7×10^{-5}	2×10^{8}
Sr-90	28.	2.3×10^{4}	8×10^{-7}	3×10^{10}
Y-91	0.159	2.3×10^{5}	3×10^{-4}	8×10^{8}
Zr-95	0.178	2.7×10^{5}	6×10^{-4}	4×10^{8}
Ru-103	0.110	3.5×10^{4}	9×10^{-4}	4×10^{7}
Ru-106	1.0	7.2×10^{4}	1×10^{-4}	7×10^{8}
Te-129	0.090	3.1×10^{3}	2×10^{-4}	2×10^{7}
Cs-137	30.	2×10^{4}	2×10^{-3}	1×10^{7}
Ba-140	0.035	5.2×10^{3}	3×10^{-4}	2×10^{7}
Ce-141	0.088	9×10^{4}	4×10^{-1}	2×10^{5}
Pr-143	0.038	5.5×10^{3}	5×10^{-4}	1×10^{7}
Ce-144	0.78	6.9×10^{5}	1×10^{-4}	7×10^{9}
Nd-147	0.032	1.6×10^{3}	6×10^{-4}	3×10^{6}
Pm-147	2.6	9×10^{4}	2×10^{-3}	4×10^{7}
Sm-151	80.	5.6×10^{2}	8×10^{-3}	7×10^{4}

(1) Assumes the fission products from one ton of irradiated uranium (2500 MWD/T at 5 MW/t) are segregated in 100 gallons of water ninety days after reactor discharge.

(2) The maximum permissible concentration (MPC) for each nuclide in potable water is given for each parent in equilibrium with its radioactive daughters. These values were taken from HW-25457, Rev. 1, Manual of Radiation Protection Standards, Hanford Atomic Products Operation, May 1, 1957.

Self-Heating Characteristics

The decay of the fission products creates heat as the beta and gamma emissions are absorbed in the surrounding mass. In Figure 1, the rate of heat evolution from the decay of fission products in the hypothetical ton of uranium described above is presented as a function of time. This hypothetical situation is extended in Figure 2 to depict the operating situation in which one ton of this uranium is processed per day after ninety days cooling, and the radioactive waste from this operation accumulated in a storage tank for different periods of time.

The operation of a waste storage tank is divided into two periods. The first, or filling period, is the time during which fission products are being added to the tank; the second, or cooling period, occurs after the filling operation terminates. During the filling period, the major portion of the heat generated is from the decay of short-lived fission products. The total rate of heat generation in the tank rises rapidly at first, the rate of increase being roughly proportional to the quantity of short-lived fission products added to the tank. In a few months, however, the short-lived fission products which were added to the tank in the first increments have decayed such that the cumulative rate of decay of the contained fission products approaches the rate of addition of radioactivity in the fresh wastes, and the rate of heat evolution becomes essentially constant.

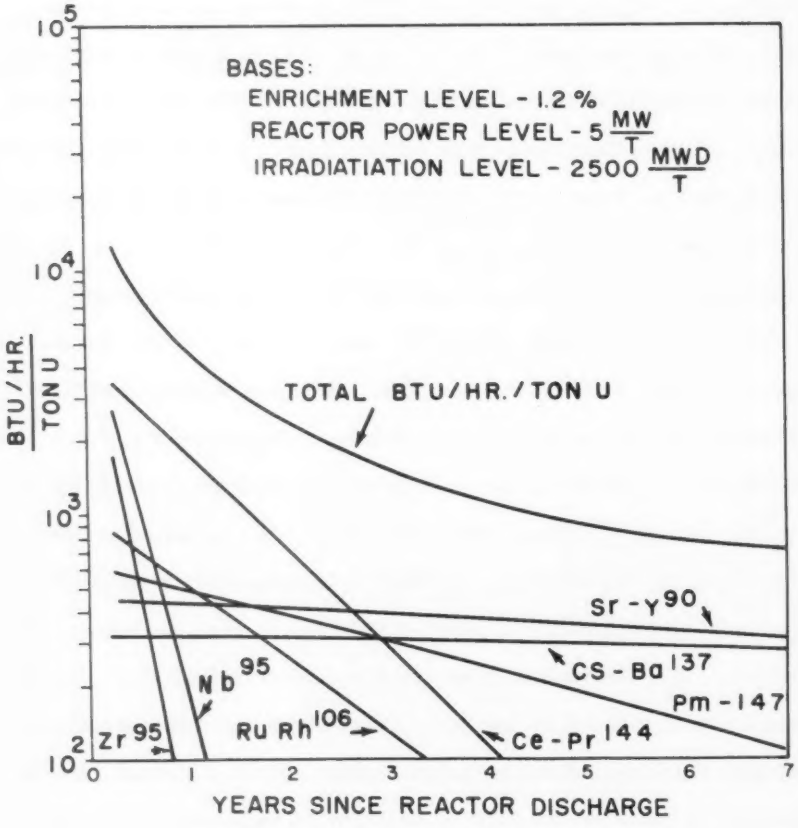

FIGURE - 1
HEAT GENERATION IN SLIGHTLY ENRICHED URANIUM
TOTAL & SOME SPECIFIC FISSION PRODUCT CONTRIBUTIONS

The cooling period curves, shown in Figure 2, represent the rate of heat generation which would be observed in a tank if no more fission products are added after the indicated periods of time. A tank filled for a short period of time contains a relatively large fraction of the short-lived fission products and the rate of heat generation falls rapidly. A tank filled for a much longer time, however, contains a larger fraction of the long-lived fission products and the rate of heat generation diminishes much more slowly.

The Storage Facilities

As shown in Figure 1 of Part 1, the separations are of the Hanford plant is located on a sandy plateau several miles from the nearest surface water-way or inhabited area. The ground level is 200 feet or more above the water table in this area. The underground waste storage tanks were constructed in groups of six to eighteen, called "farms". Surrounding each farm are two or three wells which penetrate to groundwater, and five to ten dry-wells 100 to 150 feet deep. Table 4 presents some general data concerning the sizes, costs, and time of construction of the waste tank farms at Hanford.

The pertinent features of a typical tank are shown in Figure 3. Most of the tanks are 75 feet in diameter, with the depth available for liquid storage varying from 16 to 30 feet as required to obtain the indicated volume. The tanks are constructed of reinforced concrete. A carbon steel liner covers the bottom and sides of the tank, the concrete dome being exposed to the process vapors. Pipe risers

(For figure 3, referred to above, see exhibit 10, p. 193.)

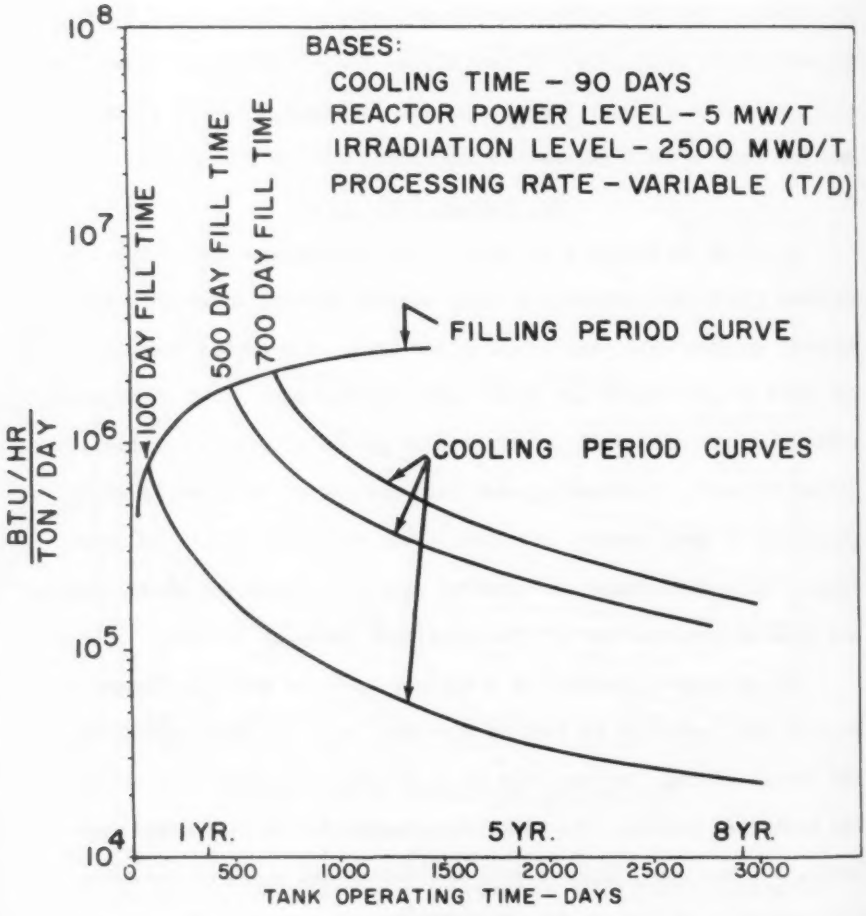

FIGURE – 2
HEAT GENERATION
IN
SEPARATIONS WASTE STORAGE TANKS

Table 4

General Information on Waste Storage Tanks at Hanford

Farm	Tanks/ Farm	Capacity/Tank (Gallons)	Capacity/Farm (Gallons	Year Constructed	Cost/Farm*	Cost/Gallon
T	16	54,500(4) 530,000(12)	6,617,000	1943-44	$3,087,000	$ 0.467
U	16	54,500(4) 530,000(12)	6,578,000	1943-44	2,969,000	0.451
B	16	54,500(4) 530,000(12)	6,590,000	1943-44	3,019,000	0.458
C	16	54,500(4) 530,000(12)	6,578,000	1943-44	2,938,000	0.447
BX	12	530,000	6,360,000	1946-47	2,208,000	0.347
TX	18	758,000	13,644,000	1947-48	5,859,000	0.429
BY	12	758,000	9,096,000	1948-49	2,651,000	0.291
S	12	758,000	9,096,000	1950-51	3,961,000	0.435
TY	6	758,000	4,548,000	1951-52	1,846,000	0.406
SX	15	1,000,000	15,000,000	1953-54	3,983,000	0.266
A	6	1,000,000	6,000,000	1954-55	4,989,000	0.831
Totals	145		90,107,000		$37,510,000	$ 0.416 Average

*Includes original tank farm cost plus improvements, including instrumentation, agitation systems, and transfer lines from separation plants. "A" Farm costs include the estimated cost of remotely-maintained stainless steel condensers and auxiliaries ($800,000) currently being installed. Operating and maintenance costs of approximately $200,000 per year for the entire complex is not included.

connect the tank with the ground surface, and provide entry for the addition of instruments or chemicals as required. Air driven recirculators are provided if the tank contents are expected to self-boil, in order to minimize problems of heat dissipation. These units are comprised of mild-steel draft tubes 24 inches in diameter, with an inverted-funnel-type entrance suspended a few feet above the bottom of the tank. The surface of the liquid is kept a few feet or more above the top of the draft tube. Air is introduced at several points near the bottom of the tube and rises through the tube. The decreased bulk density of the contents of the tube causes the liquid to rise, thereby establishing a recirculation of liquid in the tank. A recirculation rate of several hundred gallons per minute can be established with an air flow of a few cubic feet per minute. Standby air compressors are maintained to assure that the tank contents can be recirculated as required.

In Figure 4, the relationship of a given tank to the over-all waste storage facility is presented schematically. Wastes are added to the tanks through underground lines. Diversion boxes permit operating flexibility through the use of remotely replaceable pipe jumpers, whereby any given tank can be filled from many different sources. Vapors from self-boiling wastes are routed through headers to de-entrainment devices and condensers, the condensers being vented to the atmosphere through glass-fiber filters. The older tanks designed for non-boiling wastes were vented directly to the atmosphere through ground-level nozzles.

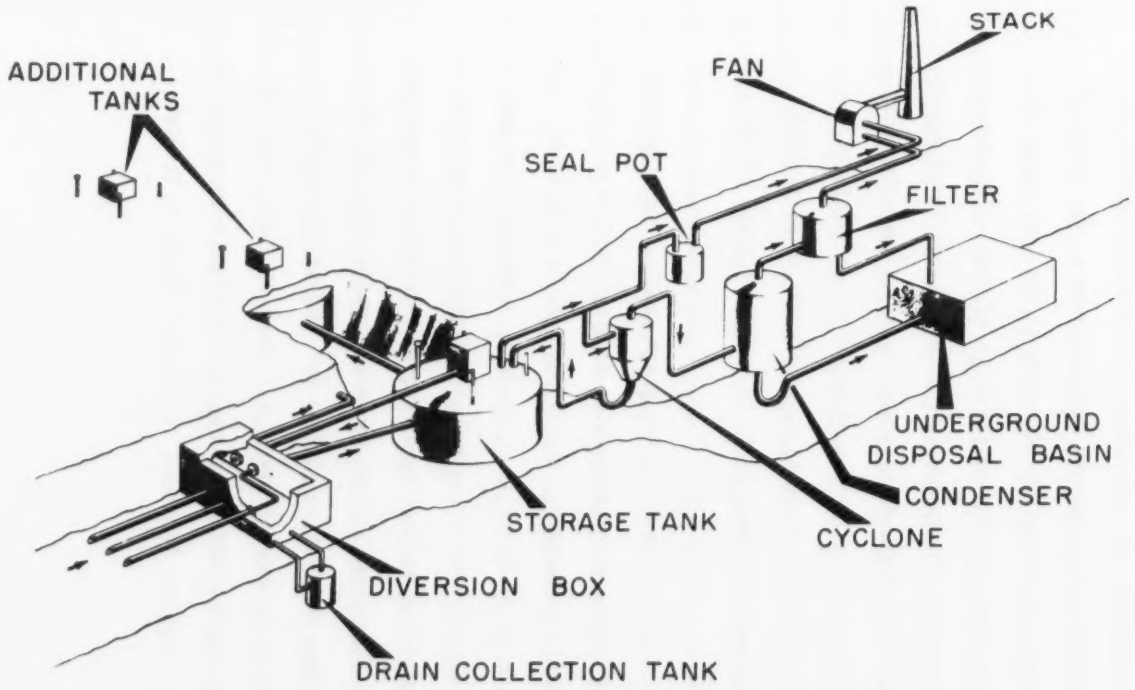

FIGURE – 4
TYPICAL HIGH LEVEL WASTE STORAGE FACILITY

Management Philosophy

The problem of permitting a significant quantity of fission products to reach the environment was recognized before operations were started at Hanford, and our operations have always been planned on the premise that such a release must not occur. The design of each waste handling facility has incorporated the best technology available at the time, and the construction of the facilities carefully inspected to assure adherence to material specifications and quality workmanship. Whenever possible, proposed operating conditions are simulated in the laboratory to define and evaluate the problems involved. Pilot plant tests supplement these studies, providing additional assurance that the plant operation can be conducted safely.

Despite the laboratory and pilot plant studies, many phenomena must be studied in the plant facilities to obtain a complete picture of the controlling parameters. The storage of the high-level wastes is therefore conducted as an in-plant development program. Most of the operating limits are maintained at values consistent with those proved safe by past experience. In those areas requiring additional knowledge, revised procedures and operating limits are developed from engineering studies and temporarily approved for test by operating and technical management. Within these temporary limits, storage conditions are then modified under continual observation by competent technical personnel. Based on the data

thus obtained, the standard operating procedures and limits are periodically revised, and new tank designs and operating philosophies are evolved in an orderly manner.

Operational Experience and Technology

Approximately 52,000,000 gallons of radioactive wastes are currently being stored at Hanford, representing the cumulative volume of wastes from many operating programs. Starting in 1944, wastes were stored from the operation of the bismuth phosphate precipitation process for the recovery of plutonium from irradiated uranium. Most of the uranium and fission products were stored in the $BiPO_4$ high-level (metal) wastes, while smaller quantities of fission products were stored in the wastes from the first and second decontamination cycles. Some of the wastes containing minor amounts of fission products were later routed to the ground, and the rest were concentrated in a waste evaporator constructed for this purpose. Later the metal wastes were reworked for the recovery of uranium. In 1952, a scavenging process was developed to reduce the concentration of dissolved cesium and strontium in the residual wastes from the uranium recovery operation. This process was used to reduce the volume of stored wastes by routing the supernates from the scavenging process to the ground and retaining the precipitate as a high-level waste. As the Redox and Purex processes were subsequently placed in operation and improved, the waste volumes per ton of uranium processed were progressively reduced.

This trend resulted in a progressive increase in the degree of self-heating experienced, and self-concentration of wastes became a reality.

The storage of these wastes has generally been routine, with stable inventories and contamination-free soil and ground waters attesting the containment of the wastes. The few unusual observations were investigated and preventive action taken as required to protect the environs. In January, 1956, for example, unusual fluctuations were recorded on instruments indicating the contents of a tank. It was found that the bottom steel plate had buckled upward, and a suspected tear was observed by periscopic inspection. Subsequent inspection failed to verify the tear, however, and the tank was later filled with water which remained at a constant liquid level. In 1957, a steady loss of liquid at a rate of 50 to 100 gallons per day was observed in another tank. This loss was first observed immediately after the vessel vent system had been revised, and it was found that air was being swept through the vapor space of the vessel into the farm condenser. The wastes were self-heated to about 135° F, and it was theorized that the loss was due to evaporation. This condition was verified by preventing the air-sweep, after which the liquid level remained constant over a three-week test period. Cause for concern again developed in May, 1958, when the bottom steel plate of a tank was found to be bowed upward as much as four feet. The vessel was being filled

with wastes from the Redox process, and the wastes were self-concentrating at a low rate. In view of the questionable integrity of the tank, the self-boiling wastes were transferred to another tank. The bottom steel plate subsequently receded to approximately its original position, although it continued to shift for several weeks before stablizing. No contamination in the ground has been detected in monitoring wells in the vicinity nor in five additional probes installed directly beneath the tank. While the intense radioactivity and high temperatures in the tank have so far precluded visual observation of the steel liner, efforts are continuing to determine the sequence and causes of the unusual events.

A safety factor of large but indeterminate magnitude has been recognized in the use of underground tanks to contain fission product wastes at Hanford. If a tank should leak, the two hundred feet of dry soil beneath the tank would retain several tens of thousand gallons of waste by capillarity before permitting the liquid to enter the ground waters. In addition, experiments with non-radioactive materials indicate that the solids present in the wastes would so reduce the transmissibility of the soil that a leak would be essentially self-sealing. These safety factors have not been relied upon, since no leaking tank has yet been discovered.

Heat Dissipation

Operating data indicate that heat is lost from a typical Hanford underground tank at rates up to 1,000,000 B.t.u. per hour

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as the tank is heated to the boiling point. Later, as the soil becomes thoroughly dried and the heat loss reaches a steady state, the rate of heat loss is reduced to the range of 200,000 to 500,000 B.t.u./hr. It is believed that essentially all of the heat lost to the ground from the Hanford tanks then escapes to the atmosphere.

When the rate of heat generation exceeds the rate of heat dissipation through the soil, the temperature of the entire mass increases until the aqueous solution starts to boil. Self-concentration then proceeds at rates up to several gallons per minute. The steam so produced is condensed in water-cooled condensers. This condensate is returned to the tank or routed to the ground, depending on the specific operating conditions. Construction work is in progress to return the most radioactive of these condensates to a separations plant for additional decontamination prior to disposal.

Self-Boiling Phenomena

Large masses of simmering liquid tend to release heat at an uneven rate. Shallow pools of solution (say, a few feet deep) tend to boil gently and self-concentration proceeds uniformly. Deeper pools of simmering liquid, however, periodically boil at an accelerated rate up to 50 to 100 times the average rate of evaporation, subsiding to normal rates in 20 to 30 minutes. This "bumping" action is believed to result from the rapid release of heat which has been stored in the liquid deep in the tank, where

the boiling point of the solution is increased by the hydrostatic head upon it. The rapid release may be triggered by any phenomenon which would start a local current of solution upward, such as the insertion of a foreign object into the tank, by self-induced movements within the tank, or by other mechanical movements. The rising liquid is subjected to a decreasing hydrostatic pressure, permitting it to boil. The vapor thus formed further reduces the pressure on the rising liquid, thereby accelerating the boiling action. The rapid boiling then continues until the reservoir of heat has been dissipated to the extent that the local upward movement of the liquid is no longer self-energized.

These phenomena have been studied extensively, both in pilot plant systems and in the stored waste systems. The maximum pressures developed within a given tank by this rapid boiling action are believed to be determined by the pressure drop in the vapor-handling system, the depth of liquid in the tank, and possibly other operating variables. Up to the present time, the maximum vapor pressure observed in any tank has been about 2 pounds per square inch, and the maximum heat released in a given "bump" about 5,000,000 B.t.u. In some of the vessels, a series of five to twenty-five rapid pressure fluctuations was sometimes observed, followed by the normal gradual decrease of heat release rate. These pressure fluctuations typically had a cycle time of about

thirty seconds, with the pressure increases developing during one to twelve second intervals.

While bumping is now prevented by the operation of air-lift recirculators, some assurance is required that the tank structure will not be damaged by pressure surges if they occur. Conservative operating limits have been established, based on the strength of the specific tank structures and the maximum pressure theoretically possible under bumping conditions. The maximum gas pressure theoretically attainable in a given tank is equal to the liquid pressure normally exerted at the bottom of the tank. This relationship exists because an increment of liquid, suddenly transported from the bottom to the top of a liquid pool, can exert a vapor pressure no greater than the hydrostatic head to which it was subjected prior to turnover; otherwise the liquid would boil in place. The dome of the buried Hanford tanks is capable of containing a sustained gas pressure of about ten pounds per square inch. An analysis of the dome structure indicated that the gas pressure would have to be developed in a small fraction of a second before a significant allowance would be required for impact loading. The dome cannot be over-stressed by pressure, then, if the hydrostatic head at the base of the tank does not exceed ten pounds per square inch. This "hydrostatic head limitation" was placed on the amount of self-concentrating wastes to be stored in a given vessel until further development work could establish a basis for the safe

use of more of the tank volume. Experimentation is continuing to obtain a better understanding of the bumping phenomena and the controlling parameters.

Temperature Control

Some control of the temperature of the decaying wastes is required to assure that the concrete tank does not lose strength through thermal stress or dehydration, and that the nitrate and nitrite salts are not heated to the point of calcination. The thermal decomposition of nitrate and nitrite salts can be expected at temperatures above 450° F, and the dehydration of concrete also becomes significant at these temperatures.

Thermal stresses within a reinforced concrete wall can potentially damage the structure if a large temperature gradient is established through the wall. If, for example, the inside of the concrete tank wall is heated to a considerably higher temperature than is the outside, the inside reinforcing members undergo compression while the outside members are placed in tension. Under these conditions, the strength of the wall is considerably less than that under isothermal conditions, and the structure could conceivably be weakened by the internal stress. The internal stresses within the tank walls can be held to reasonable limits if the thermal gradient across the wall is held to about 60° F across the two-foot thick wall. These potential stresses are minimized in Hanford tanks by partially filling the vessel with

water prior to adding the self-heating wastes, thereby reducing the rate of temperature rise at any point in the tank. About six months before a given vessel is required for the storage of current wastes, the vessel is preheated to about 150° F by adding a limited volume of process wastes. The tank and surrounding soil are thus heated gradually with small thermal gradients being established in the tank walls. Under these conditions, the tank walls expand gradually against the surrounding soil, and some additional strength is probably obtained by the restraining action of the soil.

As indicated previously, most of the heat is evolved in the vicinity of the flocculent precipitates of hydrous oxides of iron and aluminum which adsorb most of the heat-producing fission products. So long as these materials are kept suspended, the heat is readily dissipated by the evaporation of the water. If, however, the solids are permitted to accumulate on the bottom of the tank, the heat must escape the sludge layer by conduction. Since the thermal conductivity of a sludge layer is relatively low, high temperatures may be created. The maximum temperature attainable by a given system is a complex function of the quantity and "age" of the fission products present, the nature and quantity of the chemical precipitates, and the degree of agitation. The maximum temperature observed in an agitated self-concentrating system is normally 275° - 300° F, although somewhat higher temperatures have infrequently been experienced during brief periods of inhibited agitation. It is predicted

that the necessity of agitating the more concentrated wastes will be limited to a few years, after which the short-lived isotopes will have decayed to innocuous levels and the heat from the decay of the longer-lived isotopes will be able to escape at acceptable temperatures.

Chemical Instability

Water and salts are slowly decomposed by radiation under the conditions of waste storage. The decomposition of water produces hydrogen and oxygen, an estimated 40 standard cubic feet of hydrogen being evolved for every million B.t.u. of heat released. So long as the stored wastes are boiling at a significant rate, the hydrogen is continuously swept from the vessel by the steam which also acts as an effective inert gas blanket. Non-boiling tanks are vented in such a way as to permit the hydrogen to escape as it is formed. Analyses of the atmosphere within boiling and non-boiling Hanford tanks have revealed hydrogen concentrations (on a water-free basis) less than one-tenth the concentration of its lower explosive limit.

The radiation breakdown of sodium nitrate to form sodium nitrite and oxygen is a well known phenomenon. The concentration of nitrite ion in the Hanford tanks approaches an equilibrium concentration of about three molar, when the total sodium ion concentration is about eight to nine molar. The nitrite ion is decomposed further to form nitrous oxide, nitric oxide, and nitrogen dioxide. The evolution of

nitrogen oxides is normally very slow, but is accelerated at higher-than-normal temperatures. The alklaine liquid in the tank reacts with any nitrogen dioxide formed and thereby protects the tank liner from chemical corrosion by the humid nitrogen oxide system. This condition was verified by installing a finger condenser in the vapor space of the tank, and determining that the condensate formed is alkaline. The non-condensibles passing the plant condenser react with oxygen and water to form nitric acid, however, and significant corrosion has been experienced in the carbon steel condenser, vent, and vent drain systems. This equipment is now being replaced with stainless steel equipment.

Life Expectancy

The chemical systems being stored are essentially inert to carbon steel, and an extended life can be predicted for the tanks provided the operating conditions are controlled to minimize the chemical and thermal stresses discussed above. Laboratory data indicate that the carbon steel exposed to simulated neutralized Purex waste media at 220° F corrodes at a rate between 10^{-5} and 10^{-4} inches per month. These laboratory tests are being supplemented at Hanford using test specimens suspended in the self-boiling plant wastes. The experimentation is tedious and expensive, and the results are suspect because extensive decontamination work is required before the coupons can be observed. So far, the plant test results qualitatively support the laboratory data. Additional reassurance is gained from periscopic observations of tank liners after ten years of service in the storage of self-heating (but non-

boiling) wastes. The steel liner showed no detectable signs of attack; in many cases, the prime coat of paint was still intact. Based on these data, the 3/8" steel side liner in a Purex waste tank could be expected to last at least 500 years. The bottom plate could be exposed to higher temperatures and might corrode somewhat more rapidly. The liner exposed to the condensing vapors might also corrode more rapidly. Even so, a life expectancy measured in decades appears to be very conservative. If and when evidence develops to cast doubt on the integrity of a given tank, the contents would be transferred to another tank.

The corrosion tests involving the decladding wastes from future fuels are not yet complete. No unique problems are anticipated, however, and all of the data obtained thus far are similar to that reported above.

Transfer of Wastes

During the recovery of uranium from the $BiPO_4$ wastes, a considerable technology was evolved to support the transfer of radioactive solutions and slurries. It was found that commercially available multistage deep-well turbine pumps give good service in this use. Sludge layers were slurried and removed by standard sluice-mining techniques, modified as required to cope with the problems of radiation and contamination. The technology resulting from these programs was used to transfer self-boiling wastes from one tank to another in the one instance of such transfer noted above.

PART 5

DISPOSAL OF INTERMEDIATE AND LOW-LEVEL WASTES TO THE GROUND

D. W. Pearce

Water Supply

Operation of the Hanford separation plants requires four to six billion gallons of water per year. All of this water is pumped from the Columbia River at the reactor areas and is transported to the separations areas through thirty-inch diameter cross-country pipe. Four of the reactor pumping stations are interconnected and each of these is capable of suppling the additional volume required for the separation plants. The pumps, piping and associated valving and miscellaneous equipment are together termed the "export system".

A power house is located in each of the separations plant areas. Included in these facilities are steam generating units, emergency power generators and water treatment facilities. The raw water is delivered to these central facilities by the export system and receives treatment according to its ultimate use. Sanitary water is filtered and chlorinated, water for the boilers is deionized, and process water is deionized for certain of its uses.

Only about 50% of the water imported is used directly in the separations processes. The remainder is used for steam to heat the buildings, for air conditioning, human consumption, and sanitation. Much of the wastes from these operations is disposed of to the ground via the sanitary sewer systems to tile fields in each area.

Water used directly in the process becomes contaminated with radioactive materials to various degrees depending upon its use. It is used as steam for heating and evaporating radioactive solutions and also for maintaining certain reaction vessels and process solutions at elevated temperatures. It is used

in the preparation of chemical solutions, in adjusting the concentration of radioactive solutions for process reasons, and for cooling and condensing radioactive vapors from process solution evaporators and from the boiling "high level" waste storage tanks.

Only a small fraction of the total imported water leaves the separations plants in high radioactivity level waste streams to be stored in underground tanks. Most of it leaves as low and intermediate-level radioactive wastes to be disposed of directly to the ground. It is the disposal of these wastes to the ground that is to be described at this time. The ground disposal of these wastes is best described from the point of view of the facility into which they are led. Figure 1 illustrates these schematically.

Types of Ground Disposal

Swamp sites are natural depressions on the ground surface in the separations areas into which the essentially uncontaminated cooling waters* are discharged. The water seeps from the ground surface through the soils to the water table generally some hundreds of feet below.

Cribs and caverns are disposal facilities located underground to minimize radiation levels at the surface through natural earth shielding. The classical design was a box-like timbered structure while more recent types have trapezoidal cross-sections and may vary in length from 20 to 1600 feet. They are filled with washed and sized gravel to promote even distribution of the

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*In the fuel preparations and laboratory area, large volumes of water containing traces of natural and depleted uranium are disposed to ponds near the river. These solutions arise from normal plant processing and testing of fuel elements and from the pilot plant testing of proposed separations processes. The significance of this ground disposal has been discussed in Part 1 of this series.

(For figure 1, referred to above, see exhibit 11, p. 194.)

waste solution. The wastes are piped into them underground and from them the wastes percolate through the soils toward the water table. Natural ion exchange with soil minerals is utilized to retain fission products in the soil column. Any radioisotopes leaving this exchange column are detected in steel lined, perforated monitoring wells drilled around the facility and down the water table gradient.

Trenches are shallower, once-used ground disposal facilities. These receive isolated batches of waste that would interfere with ground exchange reactions if mixed with other wastes and disposed of at a common site, or wastes with high salt concentrations disallowing disposal to cribs. In the use of trenches* the disposal is based on the moisture-retaining or "specific retention" property of the dry sediments above the water table. No benefit is assumed for the ion exchange which occurs, and not more than 10% of the vertical earth column with the same cross sectional area as the trench is considered to be available.

Use of the Facilities

In order to optimize the use of swamps, cribs, and trenches and to insure the containment of the radioisotopes on the soil column, various waste streams from the plant are kept separate and not mixed with others. A primary criterion for this is the concentration of radioactive constituents present. Certain normally low-level wastes are also kept separate when there is more than a minimum potential for them to become contaminated through an equipment

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*Trench disposal is also used for smaller volumes in reactor areas and in the fuel preparation and laboratory area. In reactor areas, trenches may receive the occasional fuel element rupture debris, purge and decontamination material where soil filtration of particles is desired and, in the latter case, wastes from the removal of the jackets on rejected fuel elements and some laboratory wastes are disposed to trenches.

failure within the plant. Laboratory data and plant experience have shown that it is desirable to keep separate from each other the process cooling water, the steam condensate, and the process condensate waste streams.

The process cooling water contains less than 5×10^{-5} μc beta emitters/cc. Because of this low maximum concentration, it is disposed to a swamp. The steam condensate has, in general, somewhat higher concentrations of radioisotopes because of the higher probability of its becoming contaminated due to the failure of a heat transfer surface; it is routed to a crib. The process condensate, which is the condensed vapor from a boiling radioactive solution, is always contaminated at concentrations of about 1×10^{-1} to 1×10^{-3} μc beta emitters/cc; it is discharged to a crib.

The three individual wastes described are discharged in the plant into individual large collection pipes commonly called headers. From these the wastes either flow by gravity or are pumped through the disposal piping.

Most of the wastes that are to be discharged to the ground are transferred through stainless steel pipes located in concrete encasements buried in the ground. An exception to this is the system employed for the low-level wastes which are discharged to swamps. In this case, since radiation levels are minimal and the water is non-corrosive, above-surface black iron piping is used.

Construction of the underground piping systems is expensive and it is important to minimize any duplication. This is accomplished by terminating the main headers in diversion boxes which are centralized "valving" stations, where interconnections with existing piping may be made or from which necessary new piping may be extended.

Waste disposal to the ground at Hanford has been supported for over ten years by a closely correlated and balanced program of theoretical study,

laboratory and field experiment, and observational experience. Management of large volumes of water as well as control of the radioactive materials has been required. Extensive research and development effort in geology, hydrology, mineralogy, soil chemistry, analytical chemistry and process engineering has been applied to ground disposal problems. These will be discussed in detail in a paper prepared for the Research and Development Topic.

The geological work has defined the more significant characteristics of the underlying lava bedrock, called basalt, and the overlying sedimentary sands and clays. The hydrological work has mapped the water table (below which the earth materials are water saturated) and has followed the changes in it due to the Hanford operations. The mineralogy has proceeded hand in hand with the geology in studying the composition of the sedimentary rocks and with the soil chemistry in showing which minerals retain the radioisotopes in the sediments and the chemical mechanisms involved. Soil chemistry studies, backed by constantly improving methods of analytical chemistry, have been of major importance in that they have defined which liquid wastes may be disposed to given points in the ground and in what amount. They have also helped to define the factors which influence the ion exchange process by which the radioisotopes are held. Process engineering effort has resulted in revised plant practices which have reduced the volumes and activities of waste disposed to the ground.

It will be shown later in this description that by far the largest volumes have been disposed to swamps; lesser volumes but the greatest total number of curies have been discharged to cribs. The origin and disposal of these wastes has been described. Smallest volumes and an intermediate number of curies have been discharged to trenches. A description of the origin of these trenched materials will be of value at this point.

The first Hanford separations plants produced a high-activity level solution containing the depleted uranium and the fission products; this was placed in underground storage tanks. The uranium was later recovered from this solution and a mixed fission product solution of somewhat greater volume was left for either tank storage or ground disposal. A co-precipitating or "scavenging" process was developed to remove from this fission product solution the major parts of the Sr^{90} and Cs^{137} which were returned to the tanks as a sludge. Analysis of the solution which had been separated from the sludge and which was destined for ground disposal revealed a content of Co^{60} presumably arising from the activation of natural cobalt impurity in the fuel elements. Since the chemistry of the scavenging process rendered this radioisotope almost incapable of ion exchange in the soil (probably due to the formation of complex ions), it was necessary to dispose of most of this waste by trenching which does not rely on ion exchange as a holding mechanism.

No demonstrable hazards in the present or for the future have been created by any of these methods of disposal to the ground.

Standards

Standards on the disposal of radioactive liquid wastes to the ground are an integral part of the Radiation Protection Standards (included in the Appendix) at Hanford and these, in turn, are in accord with national recommendations on radiation protection.

There are two major problems involved in the disposal of radioactive waste to the ground. The primary one is the control of radiation exposures to within permissible limits. The second is financial. As previously described, Hanford approaches these related problems with a philosophy on waste disposal which involves controlling the discharge so that, at the point of uncontrolled uptake (which may be several miles and years away from the

initial discharge) no undue radiological problem is created. This approach allows the economic use of our detailed knowledge of the Hanford site rather than basing waste disposal practices on restrictions at the point of discharge. This approach, furthermore, requires a technical evaluation of each waste stream so that the combined effects of the entire Hanford waste output do not result in significant radiological exposures.

The most significant local standard in the area of liquid waste disposal to the ground states that "Disposal of a waste to a particular disposal facility shall cease when certain* long-lived radioisotopes are detected in concentrations of 1/10 or greater of the maximum permissible concentrations** in the ground water below the facility".

The control limit was established at 0.1 of the MPC to make allowance for possible drainage from the soil column after cut-off of the supply and to recognize the fact that the maximum concentration of contaminant in the ground water may not be detected by the adjacent monitoring wells. It is conservative since the ground disposal facilities are 7 to 10 miles from the Columbia River and any radioactive material leaving a disposal facility must filter through this distance of soil within the project boundaries before ultimate dilution in the river. Furthermore, in practice the use of a particular disposal facility is terminated before even the specified 0.1 MPC of long-lived radioisotope is detected in an adjacent monitoring well. The contaminant which controls shut-off of the facility is usually Ru^{106} which precedes Sr^{90} down the column.

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*Such as Sr^{90} and Cs^{137} with half lives exceeding three years.

**By "maximum permissible concentration" is meant the National Committee on Radiation Protection and Measurements (U.S. Bureau of Standards Handbook 52) value or the corresponding International Commission on Radiological Protection value.

Crib and trench disposal sites are fenced and access of plant personnel to them is regulated. Design drawings of the facilities and maps of the sites are maintained up to date and retained as a permanent record.

In the separations plant waste disposal practice it is an objective to retain all radioisotopes within the project boundaries so that material of any radiological significance whatsoever is not admitted to the Columbia River.

Plant Experience

Through 1958 there were about 35,000,000,000 gallons of water disposed to swamps.

The following table shows the total volumes of intermediate level wastes which have been discharged to cribs and trenches over that same time.

Volumes of Intermediate Level Wastes
Discharged to the Cribs and Trenches

Year	Annual Total gallons	Accumulated Total gallons
1944	2,000,000	2,000,000
1945	6,000,000	8,000,000
1946	10,000,000	18,000,000
1947	10,000,000	28,000,000
1948	11,000,000	39,000,000
1949	18,000,000	57,000,000
1950	20,000,000	77,000,000
1951	28,000,000	105,000,000
1952	50,000,000	155,000,000
1953	85,000,000	240,000,000
1954	770,000,000	1,010,000,000
1955	760,000,000	1,770,000,000
1956	900,000,000	2,670,000,000
1957	560,000,000	3,230,000,000
1958 (estd.)	560,000,000	3,790,000,000

Of the total shown there were over 3.7 billion gallons of water with about 1.9 million curies of gross beta-emitters disposed to the ground through a total of 71 crib structures and 28 million gallons containing 647,000 curies of gross beta-emitters disposed to 18 trench sites.

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The term "gross beta-emitters" means generally the normal spectrum of fission products and this is determined primarily by the irradiation time in the reactors and the cooling times prior to analysis. This normal spectrum is distorted in certain waste streams to a significant degree, however, by chemical processing which segregates certain of the fission product elements into different streams. The scavenging process, for example, removed important quantities of Sr^{90} and Cs^{137} and these were returned to storage in the high level waste tanks; the remaining solution was therefore depleted in strontium and cesium and, for this reason, was much less hazardous than a mixture with the normal fission product spectrum. For another example, the vapors from tanks of self-boiling waste have a composition differing greatly from the normal spectrum. These vapors, which are sent to cribs, contain a smaller percentage of Sr^{90} owing to its precipitation and retention in the tanks.

It is important to recognize too that the numbers of curies recorded above represent the magnitude of the radioactivity at time of discharge. The quantity of radioactive materials now present in the ground amounts to only about 25 per cent of that total because of radioactive decay over the years.

It is of interest to consider the approximate costs to date for the various types of disposal to the ground. Discharge to a swamp is naturally the cheapest since the cost is almost entirely that of simple piping and its installation; the facility has essentially an infinite life so unit costs at a given date mean very little. The cost per gallon disposed through 1958 is about one-hundredth cent. Disposal to cribs is the next in cost, averaging one-tenth to three-tenths cent per gallon through 1958. Trench disposal is considerably more expensive due to the severely limited volume which may be disposed to a given site; the average cost per gallon here is

three to five cents. Costs of the wells should be considered as a factor in the total cost of ground disposal. In support of this program, about 120,000 feet of wells have been drilled and cased at an average cost of about $11.50 per foot.

It need hardly be emphasized that these large volumes of wastes could not have been retained economically in storage tanks. If they had not been sent to the ground, other much more expensive means of disposal, such as evaporation, would have been required.

Adherence to the Standards

Over the years there have been three cases when isotopes with half-lives greater than 3 years have been detected in the ground water.

In 1956, Co^{60} and Cs^{137} were detected in the ground water at distances up to 1000 feet from the disposal point. At that distance the concentrations were 1×10^{-5} and 1×10^{-6} μc/cc, respectively; these were 1/40 and 1/2000 of the maximum permissible concentrations. At the present time, the Cs^{137} can no longer be detected and Co^{60} is found only immediately adjacent to the crib site; its concentration is 4.2×10^{-4} μc/cc at that point.

In 1956, Cs^{137} was detected 100 feet from a crib site in a concentration of 1.4×10^{-5} μc/cc which was 1/140 of the maximum permissible concentration. It was detected fifty feet above the water table in another well 100 feet from the crib at ten times the MPC; none was detected in monitoring wells down the water table gradient from the site. The Cs^{137} is no longer detected at this site.

In April of 1957, Sr^{90} was detected 15 feet from a crib site in a concentration of 1.0×10^{-6} μc/cc which is 1.3 times the maximum permissible concentration. None was detected in monitoring wells down the water table gradient from the crib although ruthenium, which also originated at the site,

was found up to a distance of two miles at a concentration of 1.5×10^{-7} μc/cc. The Sr^{90} concentration has continued to fall and is currently at a level of 7.9×10^{-7} μc/cc in the well adjacent to the crib.

In each of these three cases, the disposal site had been abandoned prior to the breakthrough described, no hazard arose and all evidence indicates that the long-lived radioisotopes are contained, as intended, close to the disposal site.

PART 6

GLOSSARY
RADIOACTIVE WASTE DISPOSAL
SYMBOLS AND EXPRESSIONS

Radioactive waste disposal practices utilize the knowledge of many of the natural sciences: biology, limnology, hydrology, geology, meteorology, minerology, physics, chemistry, etc., and no brief glossary can catalogue all of the symbols, terms and expressions which could be logically found in waste disposal publications. Thus the following terms are not intended to be all inclusive or rigorous in definition except in those few cases where it is judged necessary to adhere to technical phraseology.

Expressions local to Hanford are preceded by an asterisk.

Abbreviations:

alk - alkali
BTU - British Thermal Unit
c - curie (3.7×10^{10} d/s)
C - Centigrade or concentration
Co - concentration at time zero
cc - cubic centimeter
cfm - cubic feet per minute
cfs - cubic feet per second
cm^2 - square centimeter
cm^3 - cubic centimeter
*CPD - Chemical Processing Department
*CRAG - Columbia River Advisory Group
d or dis - disintegrations
D.F. - decontamination factor
d/m or dpm - disintegrations per minute

Element Abbreviations:

Symbol	Element	Atomic Number
A	Argon	18
Ac	Actinium	89
Ag	Silver	47
Al	Aluminum	13
Am	Americium	95
As	Arsenic	33
At	Astatine	85
Au	Gold	79
B	Boron	5
Ba	Barium	56
Be	Beryllium	4
Bi	Bismuth	83
Bk	Berkelium	97
Br	Bromine	35
C	Carbon	6
Ca	Calcium	20
Cd	Cadmium	48
Ce	Cerium - RE	58
Cf	Californium	98
Cl	Chlorine	17
Cm	Curium	96
Co	Cobalt	27
Cr	Chromium	24
Cs	Cesium	55
Cu	Copper	29
Dy	Dysprosium - RE	66
E	Einsteinium	99
Er	Erbium - RE	68
Eu	Europium - RE	63
F	Fluorine	9
Fe	Iron	26
Fm	Fermium	100
Fr	Francium	87
Ga	Gallium	31
Gd	Gadolinium - RE	64
Ge	Germanium	32
H	Hydrogen	1
He	Helium	2
Hf	Hafnium	72
Hg	Mercury	80
Ho	Holmium - RE	67
I	Iodine	53
In	Indium	49
Ir	Iridium	77
K	Potassium	19
Kr	Krypton	36

Symbol	Element	Atomic Number
La	Lanthanum - RE	57
Li	Lithium	3
Lu	Lutetium - RE	71
Mg	Magnesium	12
Mn	Manganese	25
Mo	Molybdenum	42
Mv	Mendelevium	101
N	Nitrogen	7
Na	Sodium	11
Nb	Niobium (Columbium)	41
Nd	Neodymium - RE	60
Ne	Neon	10
Ni	Nickel	28
No	Nobelium	102
Np	Neptunium	93
O	Oxygen	8
Os	Osmium	76
P	Phosphorus	15
Pa	Protactinium	91
Pb	Lead	82
Pd	Palladium	46
Pm	Promethium - RE	61
Po	Polonium	84
Pr	Praseodymium - RE	59
Pt	Platinum	78
Pu	Plutonium	94
Ra	Radium	88
Rb	Rubidium	37
Re	Rhenium	75
Rh	Rhodium	45
Rn	Radon	86
Ru	Ruthenium	44
S	Sulfur	16
Sb	Antimony	51
Sc	Scandium	21
Se	Selenium	34
Si	Silicon	14
Sm	Samarium - RE	62
Sn	Tin	50
Sr	Strontium	38
Ta	Tantalum	73
Tb	Terbium - RE	65
Tc	Technetium	43
Te	Tellurium	52
Th	Thorium	90
Ti	Titanium	22

Symbol	Element	Atomic Number
Tl	Thallium	81
Tm	Thulium	69
U	Uranium	92
V	Vanadium	23
W	Tungsten	74
Xe	Xenon	54
Y	Yttrium	39
Yb	Ytterbium - RE	70
Zn	Zinc	30
Zr	Zirconium	40

eq - equivalent

ev - electron volt

*FPD - Fuels Preparation Department

ft^3 - cubic feet

g - gram (454g = 1 pound)

GI - gastrointestinal (tract)

*HAPO - Hanford Atomic Products Operation

*HLO - Hanford Laboratories Operation

*HOO - Hanford Operations Office (AEC)

hr - hour

*HW - Hanford Works (document prefix)

ICRP - International Commission on Radiological Protection

ICRU - International Commission on Radiological Units and Measurements

*IPD - Irradiation Processing Department

K_a - atmospheric dilution factor

kilo - prefix meaning 1,000

l - liter

LET - linear energy transfer

m^3 - cubic meter

meq - milliequivalent

Mev - million electron volts

micro - prefix meaning 1/1,000,000 or 10^{-6}

milli - prefix meaning 1/1,000 or 10^{-3}

ml - milliliter

MPA - Maximum Permissible Amount

MPC - Maximum Permissible Concentration

MPL - Maximum Permissible Limit

mr/hr - milliroentgen per hour

mrad/hr - millirad per hour

mrem/hr - millirem per hour

NBS - National Bureau of Standards

NCRP - National Committee on Radiation Protection and Measurements

pH - index to hydrogen ion concentration; a measure of the acidity of a solution; water has a pH 7-neutral, pH values increasingly less than 7 are increasingly acidic.

ppb - parts per billion

ppm - parts per million

*PRP - Plutonium Recycle Program

*PRTR - Plutonium Recycle Test Reactor

precip - precipitated

q - a symbol for permissible body burden of a radioisotope

*"Q" - An AEC designation for security clearance; a meteorological symbol for the rate of discharge of atmospheric pollutants.

R - Generalized chemical symbol designating a constituent of minerals (calcium, magnesium etc.) and used in mineral formulas.

r - roentgen

rad - unit of absorbed dose

RBE - relative biological effectiveness

RE - rare earth

rem - unit of RBE dose; roentgen equivalent man/mammal

*RPS - Radiation Protection Standards

$T_{1/2}$ - half life, radiological or physical

T_b - half life, biological

T_{eff} - half life, effective

TBP - organic solvent (tributyl phosphate)

Z - symbol for atomic number

α - alpha particle

β - beta ray (particle)

γ - gamma ray (photon)

μ - micron or micro

μc - microcurie (10^{-6} curie)

μc/cc - microcurie per cubic centimeter

μc/gm - microcurie per gram

$\mu\mu$c - micro microcurie (10^{-12} curie)

ρ - (rho) density

Activation - Inducing radioactivity in a substance by bombardment with neutrons, alpha particles, or other particles or occasionally by gamma rays.

Activity Density - The quantity of radioactivity present in a unit quantity of matter. It can be expressed as activity mass density (μc U^{238}/gram aluminum), as activity volume density (μc S^{35}/cc water), or activity surface density (μc Pu^{239}/cm^2).

Acute Exposure - Exposure to harmful materials or radiation sustained within a few seconds to a few hours or days. (see Chronic Exposure)

Adsorption - A surface effect in which the molecules of gases, liquids, or dissolved substances cling to the surface of a solid material (distinquished from absorption).

Algae - Primitive water plants which do not have true stems, roots or leaves; most freshwater species are microscopic in size.

Alpha Particle - A positively charged particle emitted from the nucleus of an atom, composed of two protons and two neutrons and capable of ionizing matter.

Anion - A negatively charged particle or ion; collected by a positive anode. (see Cation)

Anticline - An upfolded rock mass shaped like an arch. The opposite of a syncline.

Apatite - A mineral containing calcium phosphate

Aquifer - Water-bearing rock.

Atmospheric Stability - A measure of the vertical acceleration of small volumes of air by buoyancy forces which is determined by the vertical air density gradient. In an unstable atmosphere buoyancy accelerates the motion of the air volume up or down; under neutral conditions there is no net buoyancy; under stable conditions the buoyancy forces retard the vertical motion.

Background - Ionizing radiations other than that to be measured, which contributes to electrical leakage and/or background counts. In measurements of radioactivity, the term usually refers to the presence of cosmic rays and to naturally present radioactive substances.

Basalt - A hard, dense, fine-grained rock, dark-gray to black, solidified from some lavas.

*Basin, Retention - The concrete pool or steel tank which holds reactor effluent or separations plant cooling water until it is released to the river or ground.

Beta Particle - A negative electron or a positive electron (positron) emitted from a nucleus during radioactive decay and capable of ionizing matter.

Body Burden - The total amount of radioisotope(s) deposited in the body, at the time of interest.

Bone Seeker - Element(s) having an affinity for human bone. Of particular concern in waste disposal are strontium, uranium, plutonium, radium, and several of the rare earth elements.

Breakthrough Curve (soil chemistry) - A graph used to study the ability of soil or other solids to remove radioisotopes from a flowing stream. The breakthrough point is where the first detectable or some given concentration appears in the flowing stream after passage through the medium.

Calcareous - Containing calcium carbonate, $CaCO_3$.

Capillary - Minute hair-like passage in solids into which liquid is drawn by surface tension.

Cation - That part of a chemically dissociated compound (particle or ion) having a positive electrical charge and capable of being attracted by the negative pole (cathode) of a chemical cell. (see Anion)

*Caustic Scrubber - Vessel in which a contaminated gas is washed with a solution of sodium hydroxide (caustic) to remove the contaminant. (see Scrubber Solution)

*Cavern - Gravel bed buried in the ground from which a liquid waste stream can percolate into the soil. (see Vault, Crib)

Chronic Exposure - Prolonged, continuing exposure to ionizing radiation or harmful materials for a period of weeks or years. (see Acute Exposure)

Closure - In mapping or graphing, the gap by which a contour line misses rejoining; in geology, the vertical distance between the crest of an anticline and the point of drainage (underflow) or the vertical distance between the base of a syncline and the point of drainage (overflow).

Coincidence Counting - The detection of almost simultaneously occurring nuclear events by two or more detectors. Frequently used where one detector cannot distinguish between events occurring too rapidly or where it is necessary to determine the path of a nuclear particle.

*"Cold" - Relatively non-radioactive; or requiring a minimum of personnel protection.

Concentration Factor - Generally, a number which indicates the presence of radioactivity in a substance relative to known or standard conditions. Technically, it is the ratio of activity densities. For example, radiophosphorus may be present in water in X quantity per suitable unit while for fish living in this water the radiophosphorus concentration per suitable unit may be 1000 X.

*Condensate, Process - The liquid resulting from condensing the vapors from heated or evaporated process solutions; waste process condensates are mostly water.

*Conditional Release - The provisional release of radioactively contaminated materials for future use or disposal under stated conditions.

Contamination, Radioactive - The presence of radioactive material where it is undesirable, harmful, or causes interference.

Controlled Area - A defined area in which the exposure of personnel to radiation or to radioactive material is under the supervision of a radiation safety officer. (This implies that a controlled area is one that requires control of access, occupancy, or working conditions for radiation protection purposes.)

*Controlled Injury Zone - A radiation zone wherein there are radioactive substances available to such an extent that an accident involving the injestion, inhalation or injection of foreign matter into the human body could result in a significant deposition of the radioactive substance(s).

Coolant, Reactor - The purified water (or other fluid) used to remove the heat, generated by uranium fission or radioactive decay, from a nuclear reactor.

Cosmic Rays - Radiation originating above the earth's atmosphere. This highly penetrating radiation produces ionization in air or other matter.

*Crib - Openwork box buried in the ground from which liquid waste can percolate into the soil (see Vault, Cavern); cribs have been made from timbers, concrete slats, bottomless steel tanks, etc.

Critical Organ - The ICRP defines the critical body organ as "that organ of the body receiving the radioisotope (or ionizing radiation) that results in the greatest damage to the body". In most cases it is that organ of the body that receives the greatest damage, but not necessarily, since some body organs are less essential than others to the well being of the entire body.

*Cross-Contamination - The spread of certain radioisotopes to systems where they interfere with the measurement or handling of other isotopes, e.g., plutonium in an intended uranium solution.

Curie - A unit of radioactivity defined as the quantity of any radioactive nuclide in which the number of disintegrations per second is 37 billion.

Darex - A process for the dissolving of stainless steel cladding materials in a mixture of nitric and hydrochloric acids.

Decay, Radioactive - The spontaneous change of a nucleus resulting in the emission of a particle or gamma ray. The atom containing the original nucleus is sometimes called the parent atom and the resulting atom the decay product or daughter atom.

Decay Scheme, Chain, Series, or Family - A series of radioactive nuclides in decay sequence such that each forms the next one by decay terminating in a stable, non-radioactive isotope.

Decontamination - Removal of (radioactive) contaminants.

Deflocculation (soil chemistry) - Breakup of soil clusters into smaller units or particles.

Deformed (geology) - Contorted, folded or faulted.

Diatom - Microscopic primitive water plant (alga) usually characterized by a box-like shell of silica.

*Diversion Box - An underground concrete box used to route liquid wastes between separation processing plants and storage tanks, with desired connections made by pipes (jumpers) between wall terminals.

Dose(radiological) - The quantity of radiation delivered to a specified mass or volume. The units of radiological dose have been stated by the ICRP and are based on how the measurements are performed and upon observed or postulated biological effects. The more frequently used dose units follow. Absorbed Dose of any ionizing raidation is is the energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest. The absorbed dose unit is the rad and is 100 ergs/gram. Exposure Dose is an expression technically limited to X-or gamma radiation, for which the unit is the roentgen. A roentgen is a unit of dose, defined internationally as "the quantity of X- or gamma radiation such that the associated corpuscular emission per 0.001293 grams of air produces in air, ions carrying one electrostatic unit of quantity of electricity of either sign". The RBE-dose is the absorbed dose of any ionizing radiation which has the same biological effectiveness as one rad of X-radiation with an average specific ionization of 100 ion pairs per micron of water, in terms of its air equivalent, in the same region. The unit of RBE-dose is the rem (roentgen-equivalent-man/mammal).

Dosimeter or Dosemeter - An instrument for measuring dose.

*Effluent Water, Reactor - The cooling water after it has passed through the reactor.

Electron - A sub-atomic particle having a negative electrical charge and which weighs 9.107×10^{-28} grams.

Electromagnetic Radiation - Radiation such as visible light, infrared or ultraviolet light, radio, television and radar waves, X rays and gamma rays. These radiations are electrically neutral but are made up of periodically varying electric and magnetic fields. They all travel with the velocity of light ($\sim$186,000 miles/sec).

Electron Volt (ev) - An amount of energy an electron would receive in passing through a potential difference of one volt.

Equilibrium (radioactivity) - Constant ratio of disintegration rates reached as parent radioisotope decays into daughter radioisotopes; (chemistry) - condition in which a given reaction is proceeding at the same rate as the reverse reaction; (biology) - condition in which the body intake rate or the organ deposition rate of a material is equal to the respective elimination rate.

Erg - A unit of energy. Ten million ergs released in one second is equivalent to one watt.

Exchange Capacity - A measure of the capacity of a solid to capture and hold ions from solution. (see Ion Exchange)

Film, Radiation-Sensitive - Photographic film used as a radiation monitor. It is often partially covered with small absorbers of different materials to differentiate between types and quantities of ionizing radiation.

*Film (reactor tubes) - The deposit which accumulates on the inside of the reactor tubes and outside of the fuel element jackets, which originates from the soluble and colloidal material in the coolant water and from corrosion of the aluminum tubes and fuel element jackets.

Fission, Nuclear - A nuclear reaction in which a heavy nucleus splits into two approximately equal parts, which are referred to as fission fragments or products.

Fission Products - The accumulated mass of radioactive materials resulting from fission, which consists of fission fragments and their decay products.

Fixation - The general process of fixing or holding onto a solid.

Fluviatile - Geological adjective denoting river deposited.

Fragmental Rocks - Rocks formed from consolidation of angular pieces of pre-existing rocks.

***Fuel Element** - In the Hanford reactors, the short rod of uranium fuel enclosed in an aluminum jacket. Commonly called a "slug".

Gamma Ray - Electromagnetic radiation emitted by radioactive atomic nuclei. (see X rays)

Gross Beta Emitters - A large (over 300) group of radioisotopes in which the radioactive decay process is characterized by the emission of a beta particle. The parent atom involved is called a beta emitter.

Groundwater - Water in the zone of saturation in the ground; the term generally means the regional and unconfined water body bounded on its upper surface by the water table, but in a strict sense may include also confined (artesian) water, and perched water above the water table.

Half-Life, Biological or Physical - The time required for half of a particular substance to be removed from the body or from a specified tissue by biological means, when the rate of removal is approximately exponential.

Half-Life, Effective - The time required for the amount of particular specimen of radioactive nuclide in the body or in a specified tissue to be reduced to half of its initial value, as a consequence of both biological removal and radioactive decay.

Half-Life, Radioactive - The time in which the amount of a radioactive nuclide decays to half its intial value.

Hazard - Radiation or Radiological - A widely employed term used to denote unknown or uncontrolled conditions associated with the potential or actual presence of ionizing radiation. The personnel exposure as a result of these conditions may range from potential to actual, and if actual, from extremely minor to severe.

*Header - A pipe, chamber, or manifold to which a series of tubes or connections are joined so fluid may pass freely from one to another.

*High-Level Waste - At Hanford, the radioactive liquid waste which at present is retained indefinitely in underground storage tanks. The concentration of radioactive isotopes is generally greater than 100 μc/cc.

*"Hot" - Radioactive. (see "Cold")

Hydraulic Gradient - The slope of the water table or of a river surface; the pressure head acting to cause flow, as of artesian water.

Hydrographic - Pertaining to the measurement of flow and investigation of the behavior of bodies of water, and including the surveying, sounding and charting of underground water.

Infiltration Rate - The rate of passage of water through a given surface. (see Percolation Rate)

Interbed (geology) - A bed or stratum of rock or earth material between others of a different composition.

*Intermediate-Level Waste - At Hanford, the radioactive liquid waste which usually is discharged underground when the long-lived radioisotopes are known to be retained in the soil above the water table. The concentration of radioactive isotopes is generally in the range from about 10^{-5} μc/cc to 100 μc/cc.

Ion - An atom, or aggregate of atoms, which is not electrically neutral, viz. negative ion, positive ion. In certain circumstances an electron may be described as a "negative ion".

Ion Exchange - A chemical process involving the reversible interchange of ions between a solution and a particular solid material, such as a resin or soil.

Ionization Chamber - A gas-filled enclosure containing two or more electrodes, one of which may be its wall. It is used to measure or to detect radiation by collecting the ions formed in the gas by radiation such as gamma rays, beta rays, etc.

Isopach - A line on a map drawn through points of equal thickness of a given geological formation.

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Isotope - Nuclides having the same atomic number but different mass numbers, e.g., U^{235}, U^{238} are isotopes of uranium.

Kaolinite - A type of clay.

Lava - Molten rock issued at ground surface, as from a volcano, or such rock solidified. (see Basalt)

Linear Energy Transfer (LET) - The loss of energy occurring when an ionizing particle passes through a material medium; usually expressed in kilo electron volts per micron of distance travelled (kev/μ).

Long-Lived (radioisotope) - Isotope whose half-life is long compared to that of another isotope or to a given period of time. Radioisotopes with half-lives of greater than three years are frequently classed as long-lived.

*Low-Level Waste - At Hanford, the radioactive liquid waste which is usually separations process cooling water or steam condensate, and is discharged to the ground surface. The concentration of radioactive isotopes is generally less than 10^{-5} μc/cc.

Lyotropic Series - A listing of ions in sequence of their ion exchange ability.

Mass Number - The number of neutrons and protons in a given atomic nucleus.

Maximum Permissible Concentration (MPC) - The recommended upper limit for the sustained concentration of a radioactive substance in air or water liable to enter the human body. The value of the limit is dependent among other things on the nature of the radioactive substance and its chemical form.

Metabolic Rate - The rate at which the physio-chemical processes within a living organism build-up (anabolism) and break-down (catabolism) protoplasm.

Micron - Unit of length - one millionth of a meter.

Montmorillonite - A type of clay.

Neutron - An uncharged nuclear particle of approximately the same mass as a proton. When displaced from a nucleus, it decays with a half-life of 13 minutes into a proton and an electron.

Neutron Flux - Intensity of neutrons traversing a given area ($n/cm^2/sec$).

*Niflex - A process for dissolving zirconium or stainless steel cladding materials in a mixture of nitric and hydrofluoric acid.

Noble Gas - The chemically inert elements: helium, neon, argon, krypton, xenon, radon.

Nuclear Reactor - A device in which substances containing fissionable material, e.g., uranium, with or without a moderator, are so arranged that for each nucleus undergoing fission at least one neutron is available to permit further fission and thus to render the reaction self-sustaining.

Nucleon - A constituent of the nucleus, e.g., a proton or a neutron.

Nuclide - A species of atom having a specific mass number, atomic number, and energy state.

Occupancy Factor - A "time" consideration used in establishing radiological controls such that radiation exposures under planned conditions will be within permissible limits.

Occupational - Radiation Protection has two broad exposure control categories: occupational or controlled and non-occupational or public. Occupational exposures are generally individually measured and evaluated and are limited to a dose accumulation that, in the light of present knowledge is not expected to cause appreciable bodily injury to a person at any time during his lifetime. Public exposures, for a number of reasons including the fact that individual measurements or controls are not feasible, are generally limited to 1/10 of occupational values.

*Packer, Well - A mechanical device which seals off a desired portion of a water well for detailed study.

*Palouse - Geological formation composed of wind-blown silts and sands found at Hanford.

Parent Material (radioactive precursor) - A radioactive isotope that upon disintegration yields a specific nuclide, the "daughter". For example, radium-226 is the parent for the daughter gas radon, which in turn radioactively decays through a series of elements into the non-radioactive and stable element lead (Pb^{210}).

Percolation Rate - The rate of transmission of water through a porous material.

Permeability (hydrology) - A measure of the ability of a standard area of a material to conduct water under specified water pressure conditions.

Positron - (see Beta Particle)

Power Level - The rate of heat release in a nuclear reactor, expressed usually in megawatts (million watts). Distinction must be made between thermal and electrical power levels; normally, the electrical output is about 25 to 35 per cent of the thermal power level.

Precipitate (chemistry) - Solid or particulate matter separated from a solution, normally resulting from chemical or physical changes caused by a reacting material.

*Primary Zone - Government-owned land (567 sq mi) of the Hanford project containing the principal manufacturing and laboratory facilities.

Proton - A nuclear particle having a charge equal and opposite to that of an electron and having a mass about 1800 times that of an electron.

Protoplasm - The form of matter in which animal and plant life is manifest. It is composed of complex viscous and colloidal material in water.

Pumping Test (hydrology) - A method in which groundwater is pumped from a well and the effects on the water table during drawdown and recovery are studied to characterize the aquifers.

*Purex - A solvent extraction process using tributyl phosphate (TBP) as the solvent for the recovery of uranium and plutonium from irradiated fuels.

*Purging - The scouring of nuclear reactor fuel elements and process piping by the mildly abrasive action of diatomaceous earth. Purging is used to remove accumulations of film and consequently improve the operating characteristics of the reactor.

rad - (see Dose)

Radiation, Ionizing - Electromagnetic or corpuscular radiation capable of producing ions.

Radioactivity - An atomic phenomenon in which new elements are sometimes produced through spontaneous transformations of their nuclei. The rate of this nuclear reaction cannot be altered in any known way, and thus differs fundamentally from chemical reactions which can be affected by heat, pressure, etc.

Rare Earth Elements (chemistry) - General term for fifteen elements including those listed between lanthanum and lutetium on the periodic chart of elements. They are important in waste disposal considerations because the fission of uranium generates a number of radioactive rare earth isotopes.

Recharge (hydrology) - An introduction of water into the ground resulting in raising of the water table locally.

*Redox - A solvent extraction process using ethyl isobutyl ketone (hexone) as the solvent for the recovery of uranium and plutonium from irradiated fuels.

rem - (see Dose)

*Ringold - Geological formation composed of sedimentary depositions underlying Hanford.

roentgen - (see Dose)

*Rupture, Fuel Element - Breach of the aluminum jacket containing the uranium fuel. Ruptures are caused by corrosion, pressure, mechanical damage, or other action and can result in the partial release of irradiated uranium into the reactor cooling water.

Safe - In a strict sense, all ionizing radiation is capable of causing biological damage, no matter how minute. Therefore, no radiation level higher than natural background can be regarded as categorically "safe". Thus, the word "safe", as used in radiation protection, denotes a practical exposure level that, in the light of present knowledge as evaluated by qualified experts, involves a negligible risk to any individual.

Saturated Flow - Fluid flow in which the maximum capacity is utilized; in hydrology, water flow in which the space between solid particles is filled with water (differentiated from unsaturated flow).

*Scavenging, Waste - Removal of undesirable (radioactive) ions from a solution (by absorption, adsorption, occlusion, co-precipitation, etc.) with precipitates formed by addition of special chemicals.

Scintillation Counter - A counter in which tiny flashes of light (scintillations) produced in a fluorescent material by an ionizing radiation are detected by an electronic device.

*Scrubber Solution - A liquid used to wash a gas of entrained solids, etc. in which the gas itself is only slightly soluble. (see Caustic Scrubber)

*Secondary Zone - Government-controlled land (164 sq mi) adjacent to Hanford Primary Zone, north and across the Columbia River from the reactors.

Sedimentary Rock - Rocks formed by the solidification of deposits of sediment; examples are sandstone and shale.

*Seepage Pit - Hole or depression at the ground surface into which solutions are admitted for percolation into the ground.

Seismic Method - A geophysical method used to study rock structures by recording the reflections of shock waves from controlled explosions.

Short-Lived (radioisotope) - (see Long-Lived)

*Silver Reactor - A process vessel using silver nitrate to remove iodine vapors from a gas stream.

Sludge - A wet solid settling out from a liquid. (see Supernate, Precipitate)

*Slug - (see Fuel Element)

Sparge - Force bubbles of gas through a liquid.

Specific Activity - The quantity of radioactive isotopes per unit mass of the same element or isotope associated with it.

*Specific Retention - The capacity of soil and earth to retain liquids against the force of gravity.

Spectrometer, Gamma Ray - Most radioisotopes emit characteristic gamma rays of known energy. The gamma ray spectrometer measures these characteristic rays and provides a means for quickly determining which gamma ray emitting radioisotopes are present and in what amounts.

Stable Atmosphere - (see Atmospheric Stability)

Stack - Chimney, vertical vent pipe.

*Storage Tank, Underground - A reinforced concrete tank, lined with mild (low carbon) steel and buried several feet underground, used to store highly radioactive liquid wastes.

Stratification (geology) - The natural layering of materials built up because of sequential deposition, etc.

*Sulfex - A process for dissolving stainless steel cladding materials in sulfuric acid.

Supernate - The liquor remaining after a mixture of sludge and liquid has settled. (see Scavenging, Waste)

Syncline - A geological term meaning a down-warped rock mass; a trough, the reverse of an anticline.

Target Material - Material subjected to bombardment by nuclear particles; electrons, or electromagnetic radiation. Generally, the bombardment results in the production of radioactive isotopes.

*Touchet - Geological formation composed of sedimentary glacial outwash and river deposited sands and gravels found at Hanford.

Tracer Test (hydrology) - A method of studying the rate and direction of movement of groundwater by introducing a discernible material (e. g., a fluorescent dye) into the groundwater and sampling for it at other locations.

Transmissibility - A measure of the ability of an aquifer to conduct water.

Traverse (surveying) - A series of measurements between ground points of known location.

*Trench - Open ditch into which radioactive waste (solid or liquid) is placed and then filled with dirt to prevent movement of radioactive contamination by wind erosion.

Unstable Atmosphere - (see Atmospheric Stability)

Uranyl Ion - A normal chemical form of uranium in solution.

*Vault - A large underground space, with supported roof and sides, and earthen floor, from which liquid waste can percolate into the ground. (see Cavern, Crib)

Water Table - Upper surface of the zone of saturation in the ground.

Weir - A device used to determine the rate of flow of a liquid.

X-Ray - Electromagnetic radiation commonly generated by bombarding a heavy-metal target with high speed electrons. Identical with gamma rays for human body effects.

Zeolite - Natural minerals having ion exchange properties.

*Zirflex - A process for dissolving zirconium cladding material in ammonium fluoride.

PART 7

BIBLIOGRAPHY OF SELECTED HANFORD LITERATURE ON RADIOACTIVE WASTE DISPOSAL

I. LITERATURE RELATED TO OVER-ALL WASTE DISPOSAL ACTIVITIES AT HANFORD

Smith, J. M. Jr., Nature and Source of Radioactive Wastes in Industry, Presented to: Sixth Annual Pacific Northwest Industrial Waste Conference, Seattle, Washington, March 18-19, 1954.

Healy, J. W., Waste Disposal in AEC Installation. Health Physics Conference, Ohio State University, Columbus, Ohio, pp. 24-8, June 13, 1955.

Experience at Hanford on the handling of fission product wastes resulting from the processing of uranium fuel from a nuclear reactor is discussed.

McDonald, D. W., Disposal of Radioactive Wastes at Hanford. Presented to: American Institute of Chemical Engineers Waste Symposium; North California Section; San Francisco, California, June 17, 1955.

Parker, H. M., Radiation Exposure from Environmental Hazards. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 13, pp. 305-310, 1955.

The expected exposure from both nuclear reactors and chemical processing plants are classified and discussed.

Parker, H. M., Radiation Exposure Experience in a Major Atomic Energy Facility. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 13, pp. 266-269, 1955.

The experience in exposure of personnel in in-plant operations at Hanford from 1944-1954 is summarized.

Parker, H. M. and J. M. Smith, Health Protection in Chemical Processing Plants. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 13, pp. 324-328, 1955.

Control of radiation exposure to the employees at Hanford and the elimination of deleterious effects on the environs are surveyed.

Rostenbach, R. E., Radioactive Waste Disposal at Hanford. Published in: Sewage and Industrial Wastes 28: pp. 280-6, March, 1956.

The radioactive waste disposal and control programs at Hanford are briefly outlined.

Granquist, D. P. and R. E. Tomlinson, Estimated Magnitude of Waste Disposal Problems in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955. TID-7517 (Pt. Ib), October 1956.

Rhodes, D. W., J. R. Raymond, and H. V. Clukey, Operational Experience in Hanford Liquid Waste Disposal, in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio, December 6-9, 1955. TID-7517 (Pt. II) October 1956, Confidential.

Tomlinson, R. E., Waste Disposal at Hanford. Presented to: Fifth Atomic Energy in Industry Conference; Philadelphia, Pennsylvania, March 15, 1957.

Current methods and costs of containment, together with a brief review of the problems encountered and the direction of efforts toward improvement are presented.

Hanford Atomic Products Operation, Manual of Radiation Protection Standards. HW-25457 (Rev. 1), July 1957.

Policies for control of radioactive materials and radiation exposure, radiation units and nomenclature, permissible limits, records, control procedures, radiation incidents and waste disposal are presented.

Hanson, W. C. and R. L. Browning, Role of Food Habits in Wildlife Contamination at Hanford, in Biology Research - Annual Report 1956, HW-47500, 1957.

Food items in stomachs of 522 wildfowl collected along the Columbia River and at waste ponds on the Hanford project were identified to corrleate tissue concentrations of radioisotopes with wildfowl food habits.

Pearce, D. W., Radioactive Waste Disposal at Hanford. Published in: Eighth Annual Proceedings, Pacific Northwest Industrial Waste Conference Institute of Technology, Washington State College, pp. 141-151, 1957.

The broad scientific studies which have been fundamental to liquid waste disposal technology at Hanford are discussed. Emphasis is placed on the environmental aspects of the problem.

Healy, J. W., B. V. Andersen, J. K. Soldat and H. V. Clukey, Radiation Exposure to People in the Environs of a Major Atomic Energy Plant. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958. Paper 743.

Estimates of the radiation-dosage rates to people living in an environment containing small amounts of radioactive wastes are presented. The year 1957 is used to illustrate the exposures involved.

Kornberg, H. A., Radiation Biology Research as a Supporting Function for Atomic Energy Installation. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958. Paper 390.

Radiation biology problems that arose at Hanford during the past ten years are reviewed. Their relation to production and their solutions serve to suggest how radiation biology can be a supporting function for other atomic energy installation.

II. LITERATURE RELATED TO THE RELEASE OF REACTOR EFFLUENT TO THE COLUMBIA RIVER

Parker, H. M., Status of Problem of Measurement of the Activity of Waste Water Returned to the Columbia River. HW-7-2346, September 11, 1945.

Healy, J. W., Accumulation of Radioactive Elements in Fish Immersed in Pile Effluent Water. HW-3-3442, February 27, 1946.

Foster, R. F., Some Effects of Pile Effluent Water on Young Chinook Salmon and Steelhead Trout. HW-7-4759, August 31, 1946.

> Three separate toxicity studies are discussed, namely a pilot study on chinook salmon fingerlings, an extensive study on steelhead trout fingerlings and a comprehensive study on chinook salmon eggs, fry and fingerlings.

Herde, K. E., Studies in the Accumulation of Radioactive Elements in Oncorhynchus Tschawytscha (Chinook Salmon) Exposed to a Medium of Pile Effluent Water. HW-3-5064, October 14, 1946.

> Data are presented on the accumulation of radiosodium and other longer lived elements in fish.

Herde, K. E., Radioactivity in Various Species of Fish from the Columbia and Yakima Rivers. HW-3-5501, May 14, 1947, Secret.

> This study is concerned principally with accumulation of radioactive materials by fish exposed to dilute reactor effluent in the Columbia River.

Herde, K. E., A One Year Study of Radioactivity in Columbia River Fish. HW-11344, October 25, 1948.

> Over a one year period 194 fish were collected from the Columbia River and assayed for beta-emitting materials. The results of this 12-month study are compared with data collected in earlier studies.

Herde, K. E., Radioactivity of Pile Effluent and Its Biological Significance. HW-11509, November 9, 1948, Secret.

> The fundamental characteristics of isotopes known to be present in the pile effluent are outlined and the biological effects on fish are discussed.

Olson, P. A. Jr., Some Effects of Pile Area Effluent Water on Young Silver Salmon. HW-8944, 1948.

> A detailed description is given of the second series of experiments undertaken to determine the effect of Hanford reactor effluent water on the developing eggs and young of silver salmon. The first experiments are reported in HW-7-4759.

Coopey, R. W., Preliminary Report on the Accumulation of Radioactivity as Shown by a Limnological Study of the Columbia River in the Vicinity of Hanford Works. HW-11662, November 12, 1948.

The amount of radioactivity in the bottom-living organisms of the Columbia River was investigated at nine stations in the vicinity of Hanford from November 1947 to April 1948.

Berry, E. C. and J. F. Cline, First Annual Report of the Botany Field Station. HW-16056, 1950.

The location, layout and method of irrigating the station are described. The crops grown and their yields are given. Results of analyses for radioactivity are reported. A plot of the field station and photographs are included.

Davis, J. J. and C. L. Cooper, Effect of Hanford Pile Effluent Upon Aquatic Invertebrates in the Columbia River. HW-20055 DEL, January 19, 1951.

A radiological-ecological survey of the invertebrate fauna that inhabit the Columbia River within the confines of Hanford and downstream to the McNary Dam is reported for the period October 1948 through February 1950.

Larkin, W. E., R. L. Watters, Z. E. Carey, and W. Singlevich, Preliminary Report on Radiochemical Analysis of Hanford Pile Effluent Water. HW-21844, August 1, 1951, Secret.

Honstead, J. F., J. W. Healy and H. J. Paas, Columbia River Survey Preliminary Report. HW-22851, December 14, 1951.

Data on velocity, radioisotope concentrations, and temperature measurements of the Columbia River are reported and interpreted.

Cline, J. F. and J. W. Porter, The Second Annual Report of the Botany Field Station. HW-20999, 1951.

Data are presented which show that during 1950 the radioactivity of several types of crops, irrigated with water pumped from the Columbia River below the Hanford reactors, differed but slightly from that of similar crops grown in a control area.

Pilcher, G. E., A Statistical Analysis of Measured River Contamination at Selected Shoreline Locations. HW-23642, February 27, 1952, Secret.

The relationship existing between reactor power levels, Columbia River flow and measured amount of beta-particle emitters downstream is defined and discussed.

Parker, H. M., Permissible Limits for Release of Reactor Effluent to the Columbia River. HW-24356, May 6, 1952, Confidential.

The background of the problem of determining the adequacy of the waste disposal system is reviewed and a set of recommendations providing the necessary safeguards is proposed.

Healy, J. W., The Effect of Purging During Pile Operation on Effluent Water. HW-24578, May 27, 1952, Secret.

Radiochemical analytical data on purge effluents and estimates of the hazards involved are presented. Recommendations on disposal methods and future studies are also included.

Olson, P. A., Jr. and R. F. Foster, Accumulation of Radioactivity in Columbia River Fish in the Vicinity of Hanford Works. HW-23093, July 1, 1952.

Data obtained from the radioassay of fish collected in the vicinity of Hanford between April 1948 and June 1950 are presented.

Lee, E. P., Manufacturing Department Operating Standards for the Release of Reactor Effluent to the Columbia River. HW-25167, August 8, 1952, Confidential.

Davis, J. J., R. W. Coopey, D. G. Watson, C. C. Palmiter and C. L. Cooper, The Radioactivity and Ecology of Aquatic Organisms in the Columbia River, in Biology Research - Annual Report, 1951. HW-25021, 1952.

A limnological study is presented which includes methods of sampling biota, distribution of radioisotopes in various members of the food chain, variation in activity densities of biota and population densities of benthonic forms.

Foster, R. F., Biological Problems Associated with the Discharge of Pile Effluent into the Columbia River, in Biology Research - Annual Report 1951, HW-25021, 1952.

Olson, P. A. Jr., Observations on the Transfer of Pile Effluent Radioactivity to Trout, in Biology Research - Annual Report 1951, HW-25021, 1952.

Experiments concerned with the selective concentration of isotopes from reactor effluent by fish food organisms and the transfer of some of these to trout are described.

Olson, P. A. Jr. and R. F. Foster, Effect of Pile Effluent Water on Fish, in Biology Research - Annual Report, 1951, HW-25021, 1952.

Experiments are described which demonstrate that the effect of reactor effluent on salmon and trout results from chemical toxicity and not the radioactive materials.

Porter, J. W. and J. F. Cline, The Effect of Sodium Dichromate and Hanford Pile Cooling Water on the Growth of Bean Plants, in Biology Research - Annual Report 1951, HW-25021, 1952.

Watson, D. G., Observations on the Spawning and Migration of Chinook Salmon, Oncorhynchus Tschawytscha (Walbaum) in the Columbia River in the Vicinity of Hanford Works, in Biology Research - Annual Report 1951, HW-25021, 1952.

Lutton, J. M. and W. C. A. Woods, Columbia River Water as a Pile Coolant. HW-28467, June 23, 1953, Secret.

The cooling water system of the Hanford reactors is re-examined and re-evaluated both technically and economically.

Coopey, R. W., The Abundance of the Principal Crustacea of the Columbia River and the Radioactivity They Contain. HW-25191, June 25, 1953.

The species and abundance of crustacea together with the radioactivity they contained were traced for a 14-month period.

Soldat, J. K. and G. R. Quimby, Effective Retention Time of the Hanford 107 Reactor Effluent Basins. HW-28830, August 3, 1953, Secret.

Effluent retention times for the individual Hanford reactor areas are given for 1950-1953.

Coopey, R. W., A Preliminary Comparison of the Nannoplankton and Net Plankton of the Columbia River. HW-29298, September 10, 1953.

Plankton collected from the Columbia River by the net method and the centrifuge method were compared. Similar seasonal trends were present in both types of plankton.

Coopey, R. W., Radioactive Plankton from the Columbia River. Published in: Transactions of the American Microscopic Society, Vol. 72, p. 315, 1953.

The abundance of plankton in the Columbia River and their accumulation of radioisotopes is described.

Cline, J. F., A. A. Selders and J. H. Rediske, Chronic Effects of Reactor Effluent Water on Cereal Plants, in Biology Research - Annual Report 1952, HW-28636, 1953.

The effects on barley irrigated with various dilutions of Hanford reactor effluent are presented.

Davis, J. J., R. W. Coopey, D. G. Watson and C. C. Palmiter, Radiobiological Survey of the Columbia River, in Biology Research - Annual Report 1952, HW-28636, 1953.

Radioactivity of organisms inhabiting the Columbia River is presented for the year 1952.

Foster, R. F. and P. A. Olson, Jr., Effect of Reactor Effluent in Young Silver Salmon, in Biology Research - Annual Report 1952, HW-28636, 1953.

The effects of exposing the eggs and young of silver salmon to dilute Hanford reactor effluent for 11 months are summarized.

Olson, P. A. Jr. and R. F. Foster, Extended Retention of Rainbow Trout in Dilute Reactor Effluent, in Biology Research - Annual Report 1952, HW-28636, 1953.

The effects of exposing trout to five per cent Hanford reactor effluent for a period of three years are discussed.

Whittaker, R. H., Removal of Radiophosphorus Contaminant from the Water in an Aquarium Community, in Biology Research - Annual Report 1952, HW-28636, 1953.

The efficiency with which radiophosphorous is removed from water by a mixture of organisms is described.

Wood, V. W., A Summary of Columbia River Hydrographic Information Pertinent to Hanford Works 1894-1954, HW-30347, February 24, 1954.

Wood, V. W., Hydrographic Information on Columbia River 1954-1955. Supplemental Report to HW-30347. HW-41388, March 15, 1956.

Discharge rates, gradients, and temperatures for the Columbia River are summarized on charts and tables for pre-Hanford as well as current years.

Healy, J. W., Operation of Both Sides of Reactor Effluent Retention Basins. HW-31059, March 3, 1954, Secret.

The effect of allowing the water to go directly to the Columbia River during a period when ruptured fuel elements are present is viewed from past experience.

Matsumoto, W. Y., D. L. Reid and Z. E. Carey, Report on the Radiochemical Analysis of Hanford Reactor Effluent Water for the Period January, 1951 to January, 1954. HW-32141, June 15, 1954, Secret.

The data from periodic radiochemical analyses of Hanford reactor effluent for numerous radioisotopes are reported.

Honstead, J. F., Columbia River Survey 1951-1952-1953. HW-32506, July 21, 1954, Secret.

Results of sampling for radioactivity and velocity determinations across the Columbia River at numerous locations below Hanford effluent discharge points and during various seasons are presented.

Foster, R. F., J. J. Davis and P. A. Olson, Jr., Studies on the Effect of Hanford Reactors on the Aquatic Life of the Columbia River. HW-33366, 1954, Secret.

The potential effects of reactor effluent on aquatic life of the Columbia River are pointed out and results of research work on these problems are briefly described.

Cline, J. F. and J. H. Rediske, Chronic Effects of Reactor Effluent Water on Cereal Plants, in Biology Research - Annual Report 1953, HW-30437, 1954.

In a continuation of tests started in 1952, barley was irrigated with reactor effluent. Effects on the barley are described.

Davis, J. J. D. G. Watson, C. C. Palmiter and R. W. Coopey, Radiobiological Survey of the Columbia River, in Biology Research - Annual Report 1953, HW-30437, 1954.

The relative amounts of radioactive materials found in Columbia River organisms during 1953 are reported according to species, season, and distance downstream.

Olson, P. A. and R. F. Foster, Reactor Effluent Monitoring with Young Chinook Salmon, in Biology Research - Annual Report 1953, HW-30437, 1954.

The effects of exposing the eggs and young of Chinook salmon to Hanford reactor influent and effluent for seven months are presented.

Foster, R. F. and R. E. Rostenbach, Distribution of Radioisotopes in Columbia River. Published in: Journal of American Waterworks Association, Vol. 46, Pt. 2, pp. 633-640, July, 1954.

Some of the physical and biological factors governing and resulting from the use of Columbia River water for cooling Hanford reactors are discussed.

Healy, J. W. and R. E. Rostenbach, Influence of Hanford Reactors on Domestic Use of Columbia River Water. HW-36862, May 25, 1955, Secret.

The direct effects on humans of radioactive materials in the Columbia River are reviewed. Data on the radiochemical composition and rates of discharge of reactor cooling water are reviewed for the years 1951-1955.

Rostenbach, R. E., Temperature of the Columbia River Between Priest Rapids, Washington and Umatilla, Oregon. HW-39347, October 5, 1955.

Columbia River temperature data are summarized for the period 1944-1955. Flow relationships of the Columbia and Snake Rivers are described.

Cline, J. F., A. A. Selders and J. H. Rediske, Chronic Effects of Reactor Effluent Water on Cereal Plants, in Biology Research - Annual Report 1954, HW-35917, 1955.

The effects on barley of irrigation with various dilutions of Hanford reactor effluent are described.

Davis, J. J., D. G. Watson and C. C. Palmiter, Radiobiological Survey of the Columbia River, in Biology Research - Annual Report 1954, HW-35917, 1955.

Distribution of radioactivity among Columbia River organisms for 1954 is compared to that in 1953.

Olson, P. A. and R. F. Foster, Reactor Effluent Monitoring with Young Chinook Salmon, in Biology Research - Annual Report 1954, HW-35917, 1955.

The effects of subjecting the eggs and young of chinook salmon to a series of dilutions of dechlorinated Hanford reactor influent and effluent for seven months are summarized.

Olson, P. A. and R. F. Foster, Effect of Reactor Area Effluent Water on Chinook Salmon Fingerlings, in Biology Research - Annual Report 1954, HW-35917, 1955.

Concentrations of reactor effluent which were toxic to fingerling-size chinook salmon are described.

Olson, P. A. and R. F. Foster, Effect of Reactor Area Effluent Water on Migrant Juvenile Blueback Salmon, in Biology Research - Annual Report 1954, HW-35917, 1955.

The toxicity of reactor effluent to juvenile blueback salmon was found to be comparable with toxicity to other species.

Rediske, J. H., J. F. Cline and A. A. Selders, Chromium Toxicity in Plants, in Biology Research - Annual Report 1954, HW-35917, 1955.

The effect of chromium on several functions of the bean plant is described.

Foster, R. F., Aquatic-Life Studies - Index of Radioactivity Control. Published in: General Electric Review, Vol. 58, pp. 36-39, July, 1955.

The activities of the Aquatic Biology Operation at Hanford are discussed in popularized form.

Foster, R. F. and J. J. Davis, The Accumulation of Radioactive Substances in Aquatic Forms. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 13, pp. 364-367, 1955.

Conditions found in the Columbia River below the Hanford reactors are described and an explanation is given of how aquatic forms become radioactive.

Soldat, J. K., Columbia River Travel Time Measurements by Float Methods. HW-41275, January 4, 1956.

Travel times between the most upstream Hanford reactor and the Pasco and Kennewick water plant intakes were measured by float methods. The variations in minimum travel times are discussed.

Rostenbach, R. E., Columbia River Survey - 1954-1955. HW-43529, May 10, 1956, Secret.

The temperatures and radioactivity of the Columbia River and McNary Reservoir are summarized for 1954 and 1955.

Matsumoto, W. Y., The Effect of Reactor Power Purges on Radioactive Contamination of Coolant. HW-43830, June 20, 1956, Secret.

Radiochemical analytical data from samples of purge effluents are presented. The six purges were made during reactor operation in the spring of 1955.

Davis, J. J., Concentration of Radioisotopes by Aquatic Insects in the Columbia River. Presented to: Entomological Society of America; Berkeley, California, June 26-28, 1956.

This paper discusses the concentration of radioisotopes in aquatic insects in the Columbia River. This concentration follows a seasonal pattern produced by the combined effects of the concentration of isotopes in the water, the water temperature and the physiology of the specific organism.

Olson, P. A., Effect of Sodium Dichromate in Reactor Cooling Water on Young Chinook Salmon. HW-44357, July 19, 1956, Confidential.

Koop, W. N., Influence of Purges During Reactor Operation on the Concentration of Radioisotopes in the Columbia River. HW-45125, August 31, 1956, Secret.

The effects of purges during reactor operation on river pollution are estimated and specifications for conducting purges are recommended.

Healy, J. W., Reactor Effluent Monitoring. HW-45725, October 12, 1956, Confidential.

A discussion of studies required to provide adequate monitoring of the reactor coolant and the Columbia River under conditions of proposed production increases is presented.

Healy, J. W., J. F. Honstead and W. Y. Matsumoto, Flow Patterns in Disposal to the Columbia River, in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio, December 6-9, 1955. TID-7517 (Pt. II), October, 1956. Confidential.

Davis, J. J., D. G. Watson and C. C. Palmiter, Radiobiological Studies of the Columbia River through December, 1955. HW-36074, November 7, 1956, Secret.

Data collected from September, 1948 through December, 1955 are summarized. Differences in the concentration of certain radioisotopes by various species of organisms and kinds of body tissue are described. Geographical, seasonal and annual fluctuations in the concentration of radioisotopes on organisms are discussed.

Clukey, H. V., Radiation Aspects for River Navigation Through Hanford Project. HW-47152, December 7, 1956, Confidential.

Olson, P. A. and R. F. Foster, Effect of Chronic Exposure to Sodium Dichromate on Young Chinook Salmon and Rainbow Trout, in Biology Research - Annual Report 1955, HW-41500, 1956.

Palmer, R. F., Chronic Build-Up of Reactor Effluent Radioisotopes in Rats, in Biology Research - Annual Report 1955, HW-41500, 1956.

A study is described which was designed to give a biological measure of which radioisotopes contained in reactor effluent water are the greatest potential hazard under chronic exposure condition.

Rediske, J. H., Chromium Toxicity in Plants, in Biology Research - Annual Report 1955, HW-41500, 1956.

The cause of the chlorotic effect is described. It is postulated that chlorosis is due to reduced iron uptake, thus preventing synthesis of chlorophyll.

Watson, D. G., Chinook Salmon Spawning in the Vicinity of Hanford 1947-1955, in Biology Research - Annual Report 1955, HW-41500, 1956.

Results of observations on chinook salmon spawning near Hanford from 1947-1955 are summarized and compared with the Fall run in other parts of the Columbia River.

Watson, D. G. and J. J. Davis, Concentration of Radioisotopes in Columbia River Whitefish in the Vicinity of the Hanford Atomic Products Operation. HW-48523 DEL, February 18, 1957.

Differences in concentrations of radioactive materials as related to geographical location, season, age, specific tissue and concentration of Hanford reactor effluent in the Columbia River are summarized for the period 1950-1956.

Clukey, H. V., Method of Calculating Columbia River Drinking Water Exposure. HW-49214, March 18, 1957, Confidential.

The method of calculating the theoretical exposure to humans drinking Columbia River water in the vicinity of Pasco, Washington is illustrated. The source of exposure is the radioisotopes released in the cooling water from Hanford reactors.

Norton, H. T., The Turbulent Diffusion of River Contaminants. HW-49195, March 25, 1957.

Merits and limitations of a theoretical and an empirical approach toward defining the turbulent diffusion in the Columbia River are presented.

Foster, R. F., Recommended Limit on Addition of Dichromate to the Columbia River. HW-49713, April 17, 1957, Secret.

Disposal aspects of sodium dichromate added to reactor coolant are reviewed. The recommendation is made that the monthly average concentration of hexavalent chromium in the Columbia River not exceed 0.02 ppm because of the toxic effects on important species of fish.

Ewing, R. E., Origin of Certain Radioisotopes in Hanford Reactor Cooling Water. HW-49485, April 24, 1957, Secret.

Koop, W. N., Effect of Reactor Purges When River Temperature Exceeds 15 C. HW-50601, June 10, 1957, Confidential.

The specification prohibiting purges during reactor operation, when the river temperature exceeds 15 C, is reviewed and the recommendation made that the specification be relaxed to permit a maximum of ten purges during this season.

Hall, R. B., Radioactivity in 100-Area Water. HW-52410, September 9, 1957, Secret.

A description is presented of the increased contamination of raw water supplies at downstream reactors over the years and the effects this trend has had on water purification equipment contamination and drinking water pollution at these sites.

Foster, R. F., Radioactive Tracing of the Movement of an Essential Element through an Aquatic Community with Specific Reference to Radiophosphorus. Presented to: Symposium on Marine Biological Applications on Radioisotopes; Naples, Italy, September 15-28, 1957.

This paper relates how isotopes may be used in the field to determine (1) dispersal of substances by physical, chemical and biological processes (2) availability of substances to the biota (3) modes and efficiencies of transfer of the substance between various components of the biological system (4) interrelationships between particular species and (5) seasonal changes and other ecological relationships.

Nielsen, J. M. and R. W. Perkins, The Depletion of Radioisotopes from the Columbia River by Natural Processes. HW-52908, October 4, 1957, Secret.

Pendleton, R. C., Contamination of River Organisms Grown in Retained and Non-Retained Reactor Effluent. HW-48381, November 21, 1957, Secret.

The comparative uptake by and radiation dosages to aquatic organisms grown in retained and non-retained Hanford reactor effluent diluted by Columbia River water is discussed.

Clukey, H. V., The Hanford Atomic Project and Columbia River Pollution. HW-54243 REV, December 20, 1957.

The effects of radioactive, chemical, thermal and biological pollutants from and on the Hanford reservation are discussed.

Cline, J. F., A. A. Selders and F. P. Hungate, Chronic Effects of Reactor Effluent Water on Cereal Plants, in Biology Research - Annual Report 1956, HW-47500, 1957.

Olson, P. A. and R. F. Foster, Further Studies on the Effect of Sodium Dichromate on Juvenile Chinook Salmon, in Biology Research - Annual Report 1956, HW-47500, 1957.

Palmer, R. F., Accumulation of Radioisotopes in Rats Chronically Exposed to Reactor Effluent Water, in Biology Research - Annual Report 1956, HW-47500, 1957.

A study is described which is designed to give a biological measure of the potential hazard of radioisotopes contained in reactor effluent water, under chronic exposure conditions. The phosphorus-32 concentration in bone was higher than that of any of the radioisotopes measured.

Watson, D. G., Chinook Salmon Spawning in the Vicinity of Hanford, 1956, in Biology Research - Annual Report 1956, HW-47500, 1957.

The results are summarized of the tenth annual survey of the extent of chinook salmon spawning in the Columbia River near Hanford during the Fall of 1956.

Henderson, C. and R. F. Foster, Studies of Smallmouth Bass (Micropterus dolomieu). Published in: Transactions of the American Fisheries Society, Vol. 86, pp. 112-127, 1957.

Studies on the life history of smallmouth bass within the Hanford reservation, together with their accumulation of radioisotopes from reactor effluent are described.

Krumholz, L. A. and R. F. Foster, Accumulation and Retention of Radioactivity from Fission Products and Other Radiomaterials by Fresh-Water Organisms. Published in: Chapter 9 of The Effects of Atomic Radiation on Oceanography and Fisheries, National Academy of Sciences - National Research Council Publication 551, 1957.

Olson, P. A. and R. F. Foster, Temperature Tolerance of Eggs and Young of Columbia River Chinook Salmon. Published in: Transactions of the American Fisheries Society, Vol. 85, pp. 203-207, 1957.

The ability of chinook salmon eggs to withstand temperature above those normally found in the Columbia River is presented.

McConiga, M. W. and J. K. Soldat, Predicted Effects on 100-D Area Water Supply from 100-K Area Outfall Line Failure. HW-54756, February 10, 1958, Secret.

Sampling and velocity traverses were made in the Columbia River between two of the reactor areas. The data collected were used to calculate probable beta-emitter activity densities in water of the down river reactor upon failure of the outfall lines of the upriver reactor.

Davis, J. J. and R. F. Foster, Bioaccumulation of Radioisotopes through Aquatic Food Chains. Ecology, Vol. 39, pp. 530-5, 1958.

The biological processes by which aquatic organisms pick up radioactive materials are described with the use of experience at Hanford as examples.

Davis, J. J., The Dispersion of Radioactive Materials by Streams: With Special Reference to the Columbia River. Am. Waterworks Assoc. J., Vol. 50, No. 11, pp. 1505-15, 1958.

This paper reviews the main sources of radioactive contamination of streams. Physical and chemical influences upon radioactive contaminant dispersion are discussed. Relationships between the Columbia River biota and the dispersion of certain radioactive materials in and from the river are presented.

Davis, J. J., R. W. Perkins, R. F. Palmer, W. C. Hanson and J. F. Cline, Radioactive Materials in Aquatic and Terrestial Organisms Exposed to Reactor Effluent Water. Second International Conference on The Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958. Paper 393.

The data presented emphasize the value of need for radioecological research to further elucidate and evaluate environmental relationships that pertain to radioactive waste disposal.

III. LITERATURE RELATED TO THE RELEASE OF GASES AND VAPORS TO THE ATMOSPHERE

Parker, H. M., Proposed Revision of Tolerances for I^{131}. HW-7-3042; M-4184, December 17, 1945.

> Critical examinations of tolerance values are presented. Previous values are corrected due to changes in the estimated respiration rate and in the quoted I^{131} energy.

Parker, H. M., Tolerable Concentration of Radioiodine on Edible Plants. AECD-2907; M-4183; HW-7-3217, January 14, 1946.

> Studies on the uptake of I^{131} by animals are reviewed. Data on tolerance concentration in vegetation and in drinking water are presented.

Schwendiman, L. C. and R. C. Thorburn, Size Distribution of Electrically Precipitated Particles and Isolation of Very Small Active Particles from Separations Plant Ventilating Air. HW-13074, December 16, 1948, Secret.

> Size distribution showed greater than 55 per cent of particles to be smaller than three microns.

Blasewitz, A. G., Filtration Efficiency of the T Plant Sand Filter. HW-18094, June 15, 1950, Confidential.

> Results of investigation showed presence of water resulted in lowered efficiencies.

Blasewitz, A. G., R. V. Carlisle, B. F. Judson and others, Decontamination of Dissolver Vent Gases at Hanford. HW-20332, February 15, 1951, Secret.

> Investigations are reported which relate to evolution of stack contaminants and cleanup methods leading to silver reactors for iodine removal and fiber glass filters for particulate removal.

Blasewitz, A. G., R. V. Carlisle, B. F. Judson and others, Filtration of Radioactive Aerosols by Glass Fibers, Parts I and II. HW-20847, April 16, 1951.

> This report is concerned with the intensive removal of particulate material suspended in Hanford process gaseous waste streams. An investigation of glass fibers as a filtering media is also reported.

Blasewitz, A. G. and B. F. Judson, Variation in Silver Reactor Performance. HW-21826, August 3, 1951, Secret.

> The performance of silver reactors is re-evaluated and recommendations are given.

Parker, H. M., Recapitulation of Tolerable Concentration of Radioiodine on Edible Plants. HW-25239, August 4, 1952.

The document sketches the changes in the limit and arrives at an acceptable current limit.

Blasewitz, A. G., H. A. Lee and C. E. Lapple, Sand Versus Fiber Filters. HAN-46434, August 30, 1952, Confidential.

Information gathered at Hanford on the performance of deep-bed sand filters and fibrous filters is summarized. Tentative conclusions and recommendations are given.

Wisehart, D. E. and F. E. Adley, A Study of T Plant Stack Particulates. HW-28828, July 31, 1953, Secret.

Characteristics of particulates studied are presented. Mean median particle size for all particulates and for radioactive particles is discussed.

Barad, M. L and B. Shorr, Field Studies of the Diffusion of Aerosols. American Industrial Hygiene Quarterly, Vol. 15, No. 2, pp. 136-40, June, 1954.

Experiments are described in which an aerosol generated at an elevation of 185 feet reaches the ground within a few thousand feet of the point of generation.

Fuquay, J. J. and B. Shorr, A Note on the Behavior of Stack Gases. HW-32609, December 15, 1954.

Presents basic physical concepts related to behavior of stack gases after release to atmosphere and describes geometrical features of effluent plumes related to state of the atmosphere.

Parker, H. M., Current Views on I^{131} Emission Limits. HW-36207, April 4, 1955.

The nominal daily and weekly I^{131} release limits from Hanford chemical processing plant stacks are proposed.

Schmidt, W. C., Technological Performance of Backup Silver Reactor. HW-38872, August 24, 1955, Secret.

Results of 13 tests conducted to evaluate two plant size silver reactors placed in series are described.

Fuquay, J. J., Meteorology as Related to Waste Disposal and Weapons Tests. HW-47721 A, January 15, 1957.

Meteorological factors affecting the atmospheric dispersal of radioactive particles are discussed.

Schmidt, W. C., Treatment of Gaseous Effluents. HW-49549 A, April 10, 1957.

Equipment for the removal of noxious radioactive waste gases from chemical processing facilities is described. Actual data and empirical equations for design calculations are presented.

Gast, P. F., Hanford Experience in Utilizing a Meteorological Staff in Connection with Atmosphere Pollution Problems. Presented to: Applied Meteorology Conference, Hartford, Connecticut, October 28-29, 1957.

Hilst, G. R., Some Meteorological Problems in Nuclear Energy Productions. Published in: Weatherwise, Vol. 10, No. 5, October, 1957.

This article emphasizes the meteorological factors important to the successful solution of atmospheric pollution problems and describes the meteorological program developed to cope with this problem at Hanford.

Hilst, G. R., Dispersion of Stack Gases in Stable Atmosphere. Published in: Air Pollution Control Association Journal, November, 1957.

This paper reports upon attempts to measure the diffusion and transport of materials emitted continuously from an elevated point source into stably stratified atmospheres and delineates the potential hazards and benefits which may arise from such emissions.

Fuquay, J. J., Meteorological Factors in the Control of Short-Period, Ground Level Concentrations of Separations Stack Effluents. HW-54286, December 27, 1957, Secret.

A method is developed for use in analysis and meteorological control of atmospheric pollution problems resulting from gaseous emissions from tall stacks. Special attention is given to the dilution of the gas within the stack due to addition of excess air.

Fuquay, J. J., Field Studies of Diffusion-200 Meter Scale. Transactions American Geophysical Union, 38(2): p. 269, 1957.

Experiments are described in which fluorescent material is dispersed from a ground source. Principal features of the equipment and sample analysis techniques are described. Casual relationships between some aerial dosage patterns and meteorological variables are presented.

Fuquay, J. J., Meteorological Factors in the Appraisal and Control of Acute Exposures to Stack Effluents. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 1834.

This paper purposes to clarify some of the general aspects of meteorological dispersion and control of stack effluents and to specifically develop a scheme for meteorological control of the environmental hazard or nuisance from chemically toxic materials.

IV. LITERATURE RELATED TO THE RELEASE OF RADIOACTIVE WASTES IN TANKS

Kane, J. S. and R. E. Burns, Heat Generation of Stored Redox Wastes. HW-23477, September 2, 1952, Secret.

The expected fission product activities and heat generated by fission product decay in stored wastes are calculated.

Platt, A. M., Heat Generation in Radioactive Wastes. HW-28217, July 28, 1953, Secret.

Methods are developed for calculating heat generated by radioactive decay of fission products contained in tank stored wastes.

Clukey, H. V., 300 Area Radioactive Liquid Waste Streams Disposal. HW-32762, August 20, 1954, Secret.

The three primary radioactive liquid waste streams emanating from the laboratories and fuels preparation area are discussed. Recommendations for disposal controls are also presented.

Smith, R. E. and E. A. Coppinger, Nickel Ferrocyanide Scavenging Flowsheet for Neutralized Concentrated RAW. HW-33536, October 26, 1954, Secret.

Basic processing flowsheets for the nickel ferrocyanide waste scavenging process are presented.

Bradley, J. G., Summary of TBP Waste Scavenging Performance, Economics, and Recommended Program. HW-35586, February 14, 1955, Secret.

A summary of Hanford experience in the recovery of radioactive cerium and strontium is presented. Recommendations for improving plant efficiency are made.

Smith, E. F., Structural Evaluation - Underground Waste Storage Tanks. HW-37519, June 23, 1955.

Structural analyses of the various types of underground waste storage tanks are presented. Maximum effective specific gravity and simultaneous allowable internal vapor pressure for liquid wastes at elevated temperatures are calculated.

Tomlinson, R. E., Storage of High Activity Wastes. HW-37207, July 8, 1955, Secret.

Data attesting to the soundness of storing self-heating aqueous slurries containing high concentrations of fission products in underground tanks are presented.

Anderson, C. R. and C. A. Rohrmann, The Design and Operation of High-Level Waste Storage Facilities. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 9, pp. 640-47, 1955.

The storage of high-level waste is discussed from the design and operation standpoints as related to containers, energy release, radiological safety and economics.

O'Neill, G. L. and W. H. Swift, Purex Waste Storage, Part II - 241-A Technology and Control. HW-41791 (Pt. II), October 29, 1956, Secret.

Waste farm technology is reviewed and a scheme is presented for maintaining a running inventory of the heat available from any tank.

Tomlinson, R. E. and E. A. Coppinger, Heat Problems in the Disposal of High-Level Radioactive Wastes. Published in: Chemical Engineering Progress; 52:417-21, October, 1956.

This paper discusses heat generation under operating conditions, storage of radioactive wastes in tanks, alternate disposal methods and removal of long-lived fission products.

Schulz, W. W. and T. R. McKenzie, The Removal of Cesium and Strontium from Radioactive Waste Solutions in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955, TID-7517 (Pt. Ia), October, 1956.

Methods for the removal of cesium and strontium from various aqueous waste solutions are presented.

Coppinger, E. A. and R. E. Tomlinson, Heat Problems in the Disposal of High Level Radioactive Wastes in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955, TID-7517 (Pt. Ib), October, 1956.

Van Tuyl, H. H., Recovery of Cesium from Purex Plant Wastes by Metal Ferrocyanide and Ferricyanides - II. Full Level Laboratory Investigations. HW-48830, February 28, 1957, Confidential.

Platt, A. M., The Retention of High Level Radioactive Wastes. HW-49543 A, April 18, 1957.

The retention of radioactive wastes by underground tank storage, waste scavenging, and ground disposal techniques are briefly described. Some of the engineering, economic, and biological hazard aspects of these operations are discussed.

Haney, W. A., Leak Detection-Underground Storage Tanks. HW-51026, June 20, 1957.

Data applicable in determining the significance of a leak from an underground waste storage tank, and possible leak detection methods are presented.

Platt, A. M., Heat Evaluation and Dissipation in Fixation of Radioactivity in Stable, Solid Media. TID-7550, June 21, 1957.

Calculations on heat generation and dissipation in underground tanks are presented.

Winkler, M. A. and W. E. Fry, Radiation Profiles of Purex Waste Tanks 101-A and 103-A. HW-54246, December 19, 1957.

Current status of radiation in the Purex waste tanks is given.

McKenzie, T. R., Development of a Flowsheet for the Recovery and Purification of Megacurie Quantities of Cerium-144 from Purex 1WW. HW-54500, January 14, 1958, Secret.

Haney, W. A. and J. F. Honstead, A History and Discussion of Specific Retention Disposal of Radioactive Liquid Wastes in the 200 Areas. HW-54599, January, 1958.

Stored wastes from Hanford separation processes and sites of disposal are listed. The problems of permanent retention are discussed.

Judson, B. F. and R. E. Tomlinson, Recovery of Fission Products at Hanford, Interim Progress Report. HW-55004, February 19, 1958, Secret.

Current efforts are directed toward development of a technology for cesium recovery. Studies are also devoted to cerium and strontium recovery methods.

Roberts, R. E., History and Cost of Waste Storage at HAPO. HW-55450, March 1, 1958, Secret.

History of Hanford separation process waste storage and developments which have significantly affected waste storage volume and costs since 1944 are presented.

Ames, L. L., J. R. McHenry and J. F. Honstead, The Removal of Strontium from Wastes by a Calcite-Phosphate Mechanism. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 395.

Studies of the effect of certain minerals on the ability of Hanford soils to retain cesium and strontium are described.

Pilkey, O. H., A. M. Platt and C. A. Rohrmann, The Storage of High-Level Radioactive Wastes, Design, and Operating Experience in the United States, Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 389.

Waste storage design and experience at Hanford, Savannah River, and Idaho Chemical Processing Plant are presented.

Moore, R. L. and R. E. Burns, Fission Product Recovery from Radioactive Effluents. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 1768.

Processes currently under development at Hanford for removing fission products, especially cesium, from available wastes are described.

Wilson, A. S., Tertiary Amine Extraction of Plutonium from Nitric Acid Solutions, Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 544.

A specific method for the removal and purification of plutonium from nitric acid solutions is described.

Ryan, J. L. and E. J. Wheelwright, Application of Anion Exchange to the Reprocessing of Plutonium, Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 1915.

Experiments showing that anion exchange is applicable for plutonium removal from nitric acid solutions are described.

V. LITERATURE RELATED TO THE DISPOSAL OF RADIOACTIVE WASTES TO THE GROUND

Brown, R. E. and H. G. Ruppert, The Underground Disposal of Liquid Wastes at the Hanford Works, Washington. HW-17088, February 1, 1950, Secret.

Waste disposal information and data up to 1950 are given. Hydrological and geological data obtained in the vicinity of the chemical processing plants are correlated with waste disposal practices. Conclusions on the safeness of ground disposal methods in effect are presented.

Rhodes, D. W., Preliminary Studies of Plutonium Adsorption in Hanford Soil. HW-24548, May, 1952.

Plutonium is adsorbed most effectively by the soil constituents having high surface area and high cation exchange capacity.

Healy, J. W., Release of Radioactive Wastes to Ground. HW-28121, May 20, 1953, Secret.

Past and present disposal policies and practices are reviewed. Recommendations for obtaining additional data necessary to permit major disposal policy revisions are presented.

Hanson, W. C., R. L. Browning and W. H. Braymen, Waterfowl Contamination Observed at Redox Swamp During December, 1952. HW-27104, 1953, Confidential.

The beta activity found is reported and recommendations for corrective measures are made.

McHenry, J. R., Adsorption and Retention of Cesium by Soils of the Hanford Project. HW-31011, March 2, 1954.

Basic laboratory research data concerning removal and retention of cesium from various chemical solutions by Hanford-type soils are presented.

Parker, H. M., Ground Disposal of Radioactive Wastes at the Hanford Site. HW-32041, June 4, 1954.

The Hanford environment, methods of disposal of liquid wastes, studies of radioisotope retention by soils, monitoring studies of radioisotopes in ground water and present disposal criteria are discussed.

Ruppert, H. G. and K. R. Heid, Summary of Liquid Radioactive Wastes Discharged to the Ground - 200 Areas. July 1952 through June 1954. HW-33591, October 1, 1954, Secret.

Radioactive wastes discharged to the ground in the vicinity of the Hanford Chemical Processing plants are summarized.

McHenry, J. R., Adsorption and Retention of Strontium by Soils of the Hanford Project. HW-34499, February 1, 1955.

Basic laboratory research data concerning removal and retention of strontium from various chemical solutions by Hanford-type soils are presented.

McConiga, M. W., Deformation of the Ringold Formation. HW-36373, April, 1955.

The findings of a specific geological study in the Hanford reservation are reported.

Paas, H. J. and K. R. Heid, Radioactive Contamination in Liquid Wastes Discharged to Ground at Separation Facilities through June, 1955. HW-38562, August 9, 1955, Secret.

The quantities discharged through June 1955 are summarized. Detailed data for individual disposal sites for the period July 1954 through June 1955 are presented.

Raymond, J. R., Applicability of Geophysical Methods of Exploration at HAPO. HW-38707, August, 1955.

Methods such as electric, magnetic and seismic for ground strata are discussed. Order of best choice is (1) reflection seismic (2) magnetometric and (3) dip needle.

Brown, R. E., The Feasibility of and Recommendations for an Exploratory Geophysical Program at the Hanford Works. HW-46050, August, 1955.

Paas, H. J., Unconfined Underground Radioactive Waste and Contamination in the 300 Area and Miscellaneous Areas Not Included in Other Reports. HW-39076, September 20, 1955, Confidential.

Unconfined wastes in the Hanford Laboratories Works Area and miscellaneous areas are listed.

Brown, R. E., M. W. McConiga and P. P. Rowe, Geological and Hydrological Aspects of the Disposal of Liquid Radioactive Wastes. Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955. TID-7517 Pt Ib, October, 1956.

Brown, R. E., H. M. Parker and J. M. Smith, Disposal of Liquid Wastes to the Ground. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 9, pp. 669-675, 1955.

This paper relates the generally favorable experience of ground disposal at the Hanford plant over the past decade.

Heid, K. R., Unconfined Underground Radioactive Waste and Contamination in the 200 Areas. HW-41535, January 17, 1956, Confidential.

Locations of planned and accidental discharges of radioactive wastes to the gound in the chemical processing areas are presented. Type, volume, and radioactivity of waste discharged to each site are also given.

Clukey, H. V., Recommended Limit for Radioactive Liquid Disposal from Hanford Separations Plants to Surface Ponds. HW-41440, February 16, 1956.

A maximum concentration limit is recommended for normally uncontaminated cooling water discharged to chemical processing area swamps and ponds. This limit is based primarily on wildfowl, shoreline, and windblown contamination control.

Clukey, H. V., Disposal of Radioactive Liquid Waste to Ground. Presented to: Vancouver Section, American Institute of Electrical Engineers; Vancouver, B. C., April 16, 1956.

Thomas, C. W., Cobalt-60 in Ground Water and Separations Plant Waste Streams. HW-42612, April 20, 1956, Secret.

Possible sources of cobalt-60 in scavenged waste supernatant liquid are given. Concentrations of cobalt-60 in several separations waste streams and in the regional ground water beneath the scavenged waste cribsite are presented.

Spear, W. G., A Scintillation Well-Logging System. HW-39273, May 1, 1956.

This equipment is described. It was developed to determine the extent and the energies of various gamma emitting radioisotopes in wells.

Clukey, H. V., Tabulation of Radioactive Liquid Waste Disposal Facilities. HW-43121, May 10, 1956, Confidential.

All planned ground disposal facilities, abandoned and in service, at Hanford are tabulated by operating areas. Facility location and site, and type of waste discharged to each site are also presented.

Heid, K. R., Radioactive Contamination in Liquid Wastes Discharged to Ground at Separations Facilities through June, 1956. HW-44784, August 13, 1956, Secret.

Detailed data are presented for individual disposal sites for the period July 1955 through June 1956.

Clukey, H. V. and W. A. Haney, Review of Radioactive Liquid Waste Disposal Practices for Hanford Separations Plants. HW-45058, August 24, 1956, Confidential.

Fundamental factors governing ground disposal of chemical processing plants wastes are discussed. Liquid radioactive waste disposal control and audits are also presented.

McHenry, J. R., D. W. Rhodes and P. P. Rowe, Chemical and Physical Reactions of Radioactive Liquid Wastes with Soils in Sanitary Engineering Aspects of the Atomic Energy Industry. A Seminar Sponsored by the AEC and the Public Health Service, Held at the Robert A. Taft Engineering Center, Cincinnati, Ohio, December 6-9, 1955. TID-7517 (Pt. Ia), October, 1956.

The reactions of soluble radioisotopes with Hanford soils are described. The effects of several variables are considered.

Heid, K. R., Unconfined Underground Radioactive Waste and Contamination - 100 Areas. HW-46715, November 14, 1956, Confidential.

The locations in the reactor areas where radioactive material has been discharged to the ground are listed.

Heid, K. R., Summary of Liquid Radioactive Wastes Discharged to Ground - 300 Areas Through October, 1956. HW-47055, December 7, 1956, Secret.

Wastes discharged to the ground from August 1, 1954 through October 31, 1956 in the Hanford Laboratories Works area are summarized.

Brown, R. E., Role of Geology at the Hanford Works, Washington. Published in: Bulletin Geological Society of America, Abstract 67, Vol. 12, (Pt. 2), p. 1762, 1956.

Heid, K. R., Estimated Curies of Long-Lived Beta Particle Emitters Discharged to Ground at HAPO Through December, 1956. HW-47855, January 17, 1957, Confidential.

Total curies of strontium-90 and cesium-137 present in ground disposal facilities as of December, 1956 are tabulated.

McHenry, J. R., The Removal of Cesium -137 and Strontium-90 from Scavenged Bismuth Phosphate Solutions. HW-48141, February, 1957.

Data are presented on the adsorption of cesium and strontium by Hanford soils. The volume of scavenged solutions which may safely be disposed of to the ground is determined.

Rhodes, D. W., Adsorption of Plutonium by Soil. Published in: Soil Sciences, 84, 465-71, February, 1957.

Bierschenk, W. H., Hydraulic Characteristics of Hanford Aquifers. HW-48916, March, 1957.

Geology and hydrology of the Hanford area are described and hydraulic theory is discussed. Three aquifer-performance tests are described and analyzed.

Bierschenk, W. H., The Effect of Ground-Water Mounds on the Purex Operation. HW-49728, April 18, 1957.

The direction and velocity of ground-water flow in the vicinity of the Purex plant is summarized. A method for changing the direction of ground water movement is suggested.

Rhodes, D. W. and J. L. Nelson, Disposal of Radioactive Liquid Wastes from the Uranium Recovery Plant. HW-54721, June 3, 1957.

Results of laboratory findings confirm the basis of criteria used for disposing of liquid wastes to the ground.

McHenry, J. R., Fixation of Radioisotopes in Hanford Type Soils in Fixation of Radioactivity in Stable, Solid Media, TID-7550, June 21, 1957.

The adsorption of radioactive plutonium, cerium, yttrium, cesium, and strontium by Hanford type soils is reported.

Rhodes, D. W., Waste Characteristics Governing Fixation in Soils in Fixation of Radioactivity in Stable, Solid Media, TID-7550, June 21, 1957.

The effects of variables on the ability of Hanford type soils to adsorb radioisotopes are discussed.

Raymond, J. R., A Pumping System for Groundwater Aquifer Evaluation. HW-51171, June, 1957.

A pumping system is described that permits pumping of Hanford wells to obtain data on ground water movements.

Bierschenk, W. H., Changes in the Hanford Water Table 1944 - 1957. HW-51277, July 9, 1957.

Maps showing generalized contours on the water table at Hanford indicate that throughout most of the area the general movement of ground water under natural conditions is from areas of recharge in the Rattlesnake Hills northeastward and eastward to the Columbia River.

Brandt, H. L., Calculation of Purex A-8 Crib Capacity. HW-51399, July 9, 1957.

Representative samples of Purex tank farm condensates were passed through laboratory soil columns to evaluate the radioisotope retention capacity of the crib.

Knoll, K. C., Effect of Detergents Upon Absorption of Radioisotopes in Soils. HW-52055, August 1, 1957.

Twelve detergents were tested to determine their effect upon the absorption of strontium, cesium and plutonium by soils.

Haney, W. A., Dilution of 300 Area Uranium Wastes Entering the Columbia River. HW-52401, September 9, 1957.

Sampling and analytical data are presented which show that the small quantity of uranium entering the Columbia River by ground-water flow under Hanford Laboratories Area disposal sites is diluted to background concentrations before reaching the river.

Bernard, R. M., Radioactive Contamination in Liquid Wastes Discharged to Ground at Separation Facilities through June, 1957. HW-53336, November 1, 1957, Secret.

A chronological tabulation of total beta, plutonium, uranium and significant fission product activities discharged to each Chemical Processing Area disposal facility for the period from July 1956 through June 1957 are presented.

McHenry, J. R. and J. F. Honstead, Evaluation of Sites for the Disposal of Radioactive Waste Solutions. HW-53219, November 15, 1957.

Data of soils in the vicinity of the Chemical Processing plants are presented. The data are interpreted to furnish knowledge needed to optimize selection of disposal sites.

Bierschenk, W. H. and J. R. Raymond, Hydrologic Investigations at Hanford. Published in: Transactions of American Geophysical Union, Vol. 38, 5, pp. 724-729, 1957.

Brown, R. E., Geological Conditions at Hanford Works Regulating the Disposal of Radioactive Liquid Wastes to the Ground. Published in: Transactions, American Geophysical Union, Vol. 38, 2, p. 269, 1957.

Wood, V. W., CPD Ion Exchange Capacity - Soil Resources - Process Technology - Information Report. HW-54144, January 13, 1958, Secret.

The real estate available for radioactive waste disposal by the separations plants is estimated. For Redox, a two-mile radius will permit 100 years operation. For Purex a five-mile radius will permit 90 years operation.

Brown, D. J. and R. E. Brown, A Subsurface Aeolian Deposit at the Hanford Project Washington. Presented to: Cordilleran Section, Geological Society of America. March, 1958.

An aeolian deposit of fine sand silt and clay underlies many of the waste disposal facilities at Hanford. Recognition of its presence, origin and character permitted its maximum use to assure safe disposal of wastes.

Brown, R. E. and D. J. Brown, Surface of the Basalt Series in the Pasco Basin, Washington. Presented to: Cordilleran Section, Geological Society of America, March, 1958.

The Hanford Works lies on thick basin-fill sediment underlain by generally impermeable flows of basalt. Folds in that basalt created a surface of appreciable relief that alters the direction of flow and rate of travel of ground waters and waste liquids.

Bierschenk, W. H., Hydrological Aspects of Radioactive Waste Disposal. Journal of the Sanitary Engineering Division. American Society of Civil Engineers, November, 1958, Paper No. 1835-1.

Geologic and hydrologic data have made it possible to reasonably predict the behavior of low-level radioactive wastes discharged to the ground at Hanford.

Brown, R. E. and W. H. Bierschenk, Geological and Hydrological Guides to the Ground Containment and Control of Wastes at Hanford. 4th Nuclear Energy and Science Conference, Chicago, 1958.

Geologic and hydrologic features affecting ground water and waste behavior are determined. The data obtained and principles learned are then applied to the containment and control of wastes in the ground and ground waters.

McHenry, J. R., Ionic Exchange Properties of Strontium in a Calcareous Soil. Presented to: Soil Science Society of America: Atlanta, Georgia, November 18-22, 1957. In Press: Soil Science Society of America, Proceedings (1958).

Pendleton, R. C. and W. C. Hanson, Absorption of Cesium-137 by Components of an Aquatic Community. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 392.

Results are reported of field studies and controlled experiments designed to determine if a difference exists between the availability of cesium to an aquatic community and a terrestial community.

Brown, R. E. and D. W. Pearce, et al, Experience in the Disposal of Radioactive Wastes to the Ground. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958. Paper 1767.

Twelve years of practical experience in the controlled disposal of waters to the ground at the Hanford Works and four years at the Savannah River Plant have demonstrated the feasibility, safety and economy in the disposal of at least limited volumes of some types of liquid wastes at shallow depths.

Raymond, J. R., A Shaped Charge System for Well Casing Perforating. HW-57675, October 6, 1958.

A system was adapted for nonroutine re-perforating of Hanford water wells.

VI. LITERATURE RELATED TO THE MONITORING OF RADIOACTIVE MATERIALS IN THE ENVIRONS OF THE HANFORD RESERVATION

Hanson, W. L. and R. L. Browning, Biological Monitoring, in Biology Research - Annual Report - 1951, HW-25021, 1952.

The quantity of radioisotopes found in various biological specimens of the Hanford areas are reported. Specifically, such things as radioiodine content of rabbit thyroid, radiophosphorus concentrations in waterfowl on the Columbia River and cesium and strontium concentrations in the waterfowl frequenting swamps containing fission products are reported.

Hanson, W. L. and R. L. Browning, Biological Monitoring, in Biology Research - Annual Report - 1953, HW-30437, 1954.

Hanson, W. L. and R. L. Browning, Biological Monitoring, in Biology Research - Annual Report - 1955. HW-41500, 1956.

Hanson, W. C. and H. A. Kornberg, Radioactivity in Terrestrial Animals Near an Atomic Energy Site. Proceedings of International Conference on Peaceful Uses of Atomic Energy, Vol. 13, pp. 385-388, 1955.

Results derived from a program of biological monitoring at Hanford are presented. Data show transfer of radioactive substances from the Columbia River to waterfowl, transfer of fission products from swamps to waterfowl, transfer of iodine-131 from the atmosphere to animals and transfer of radioactive particles to rabbits.

Andersen, B. V. and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period October, November, December, 1955. HW-40871, February 6, 1956, Secret.

Results obtained from monitoring the Hanford environs for radioactive contamination for a three month period are summarized. Monitoring activities include (1) radioactive contamination in effluent gases (2) radioactive contamination on vegetation (3) radioactive contamination in the atmosphere (4) radioactive contamination in Hanford waste (5) radioactive contamination in the Columbia River and related waters (6) radioactive contamination in rain and (7) radioactive contamination in drinking water and test wells. The amounts of active material discharged from plant facilities and their effect on the contamination of vegetation, air, soil and water are discussed.

Andersen, B. V. and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period January, February, March, 1956. HW-43012, May 28, 1956, Secret.

Andersen, B. V. and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period April, May, June, 1956. HW-44215, August 7, 1956, Secret.

Andersen, B. V., M. W. McConiga and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period July, August, September, 1956. HW-46726, December 7, 1956, Secret.

Andersen, B. V., M. W. McConiga and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period October, November, December, 1956. HW-48374, February 25, 1957, Secret.

McConiga, M. W. and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period January, February, March, 1957. HW-51009, June 17, 1957, Secret.

McConiga, M. W. and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period April, May, June, 1957. HW-52803, October 28, 1957, Secret.

Selby, J. M. and J. K. Soldat, Summary of Environmental Contamination Incidents at Hanford, 1952-1957. HW-54636, January 25, 1958, Confidential.

The information obtained from monitoring reports, maps and incident investigations is summarized.

McConiga, M. W., J. M. Selby and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period July, August, September, 1957. HW-54841, February 24, 1958, Secret.

McConiga, M. W., J. M. Selby and J. K. Soldat, Radioactive Contamination in the Hanford Environs for the Period October, November, December, 1957. HW-54842, February 28, 1958, Secret.

LITERATURE RELATED TO ANALYTICAL TECHNIQUES DEVELOPED IN CONNECTION WITH WASTE DISPOSAL PROBLEMS

Healy, J. W., R. C. Thorburn and Z. E. Carey, HI Control Laboratory Routine Chemical Procedures. HW-20136, 1951.

Silker, W. B., An Analytical Method for the Determination of Radioactive Cesium. HW-29959, November, 1953.

Schwendiman, L. C., Standard Practices Counting Manual. HW-30492, January, 1954.

This manual presents available data which can be used to determine disintegration rates of radioactive samples from counting rates measured under given circumstances with routinely used laboratory instruments.

Norton, H. T., P. O. Jackson and D. L. Reid, Standard Practices Radiochemical Calculation Manual. HW-46436, November, 1956.

Thomas, C. W., Radiochemical Analysis of Bravo Shot Soil Samples. HW-38987, January, 1957.

South Pacific soil samples contaminated with radioactive debris from the Bravo Test Shot were used to evaluate the transfer of radioactive elements into edible plants.

Silker, W. B., A Continuous River Sampling System and Methods of Radiochemical Analysis for Strontium-89, 90, Barium-140 and Phosphorus-32. HW-47578, January, 1957.

Equipment installed at Pasco, Washington has been in operation over a year. This equipment analyzes for strontium-89 and 90, barium-140 and phosphorus-32.

Honstead, J. F., Proportional Waste Line Sampler. U.S. Pat. 2,800,797, July 30, 1957.

Schwendiman, L. C., R. A. Harvey and H. G. Rieck, Automatic In-Line Analyzers for Radioactive Constituents in Liquid and Gaseous Waste Streams. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958. Paper 394.

Three instruments are described which have been developed for monitoring radioactive waste streams. They are (1) total beta monitor (2) gaseous waste analyzer-computer and (3) automatic waste stream analyzer.

Perkins, R. W., Gamma-Ray Spectrometric Systems of Analysis. Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1958, Paper 2377.

APPENDIX

Sections of

HAPO Radiation Protection Standards
and
Related Material

which pertain to

Radioactive Waste DisposalPractices

UNCLASSIFIED HW-25457 REV1

GENERAL ELECTRIC
HANFORD ATOMIC PRODUCTS OPERATION
RADIATION PROTECTION STANDARDS

Issued by the Manager Radiation Protection	Subject: RADIOACTIVE SOLID WASTE DISPOSAL	Classification: Radioactive Waste Disposal

I. Policy

Solid radioactive waste disposal shall be accomplished by underground burial where practicable.

II. Responsibility

1. The Operation or group generating the waste is responsible for instituting procedures that allow for the disposal of solid waste with a minimum of casual exposure to personnel.
2. The local Radiation Monitoring Operation is responsible for establishing routine and periodic survey schedules to determine that waste disposal storage and burial are undertaken with minimal personnel exposures.
3. The Manager, Radiation Protection, is responsible for being aware of the location of solid waste disposal facilities.
4. The Operating group locating new waste handling facilities should secure the prior concurrence of the Manager, Radiation Protection.

III. Standards

1. Solid radioactive waste is defined as material that is essentially dry, or whose fluids are of small volume and adequately contained.
2. Each process area should have permanently designated solid waste disposal locations and an adequate number of intermediate waste storage locations. These shall be posted as Radiation Zones.
3. Solid waste containers or packages shall be individually identified with a radiation symbol where practicable.
4. Open trenches used to receive waste shall be backfilled at a frequency sufficient to minimize the dispersal of radioactive contaminants by wind, rain, wildlife, etc.
5. Terminal conditions for direct ground disposal trenches:
   a. Backfill with a minimim of two feet of dirt cover.
   b. Surface dose rate not to exceed one mrem/hour.
   c. Permanent identification of boundaries of disposal areas, both at the site and on master diagrams and plans.
   d. Physical isolation of the disposal grounds.
6. Isolation of the basic types of contaminants should be practiced.
7. Fire, explosive and toxic by-product hazards should be considered before placing waste in a common trench.
8. Underground radioactive material, not otherwise demarcated, may be identified by signs bearing the approved radiation symbol and the word "Underground". This means should be used to designate locations where Radiation Zone conditions might be created by disturbance of the surface.

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GENERAL ELECTRIC

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HANFORD ATOMIC PRODUCTS OPERATION

RADIATION PROTECTION STANDARDS

ISSUED BY THE MANAGER RADIATION PROTECTION	SUBJECT: RADIOACTIVE LIQUID WASTE DISPOSAL	CLASSIFICATION: Radioactive Waste Disposal

GENERAL

The object of a radioactive liquid waste disposal program is economical storage or dispersion of the wastes in such a manner as to insure safety of the human population, preserve plant and animal resources, and minimize restrictions on the use of land and water.

For the purpose of this standard, radioactive liquid waste is defined as a solution, suspension, or "sludge" of radioactive material in aqueous, organic or metallic liquid, which is intended for unconfined disposal into or onto the ground or into a river.

DISPOSAL CRITERIA

1. Each waste, whether batch, intermittent or continuous stream, shall be individually evaluated, together with the associated disposal facility. As a corollary, if a disposal facility designed for one waste is to be used for other wastes, then the effect of the combined wastes must be evaluated.

2. The characteristics of the waste, as to content of radioactive and non-radioactive materials, physical composition, volume and flow rate shall be defined. The limit of variations of each characteristic shall also be defined.

3. Radioactive liquid wastes discharged to the Columbia River shall not cause activity densities of individual radioisotopes in water used for human consumption in excess of the limits specified in Radiation Protection Standard 3.1. In practice, the limits for the waste at the point of release to the river should be further reduced to allow for non-uniform dilution in the river, concentration in aquatic life, irrigated crops and domestic and wild animals, and multiple sources of waste entering the river. An allowance may be permitted for wastes discharged into the ground near the river if solids in suspension are filtered out by the soil or if radioisotopes of concern are adsorbed in the soil.

4. Radioactive liquid wastes discharged into the ground are limited in consideration of the future appearance of radioisotopes in the Columbia or Yakima rivers, the possible release of portions of the Hanford project to public use, and future expansion of plant facilities. Waste containing concentrations of radioisotopes greater than the limits specified in Radiation Protection Standard 3.1 may be approved for ground disposal where adequate soil adsorption has been demonstrated or if the liquid will be retained in the soil column for sufficient time to insure decay to the adjusted limits when the radioisotopes reach the river. Disposal of a waste to a particular disposal facility shall cease when certain long-lived radioisotopes are detected in concentrations of 1/10 or greater of the maximum permissible concentrations (Appendix A) in the ground water* below the facility.

* Ground water is defined as the regional and unconfined ground water table.

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5. Liquid wastes slightly contaminated or which potentially may be slightly contaminated with radioactive material may be approved for disposal onto the ground or into open pits. Disposal shall cease when the site constitutes an uncontrolled radiation or contamination problem.

6. Radioactive liquid waste which is immiscible with water or tends to float or sink in water requires special consideration.

INVENTORY

An adequate inventory of all radioactive material disposed of to the environment as liquid waste shall be maintained at HAPO. An adequate inventory is defined as one in which the statistical uncertainties of sampling and volume or flow measurements are consistent with the required analytical accuracy. The volume of each waste, date or period of discharge, source, disposal facility and analytical results are to be appropriately reported to the Hanford Laboratories Operation within twenty (20) days after the end of each month that the disposal facility was used. Analytical results for radioisotopes are to be reported in units of absolute quantities disposed of per month (curies or μc per month) and units of average absolute concentration (μc/cc).

RESPONSIBILITIES OF ORGANIZATION ORIGINATING WASTE

The organizational component assigned "user" responsibility for the plant facility originating radioactive liquid waste is responsible for performing, or having performed the following:

1. Determining characteristics of existing and new waste streams and appropriately reporting such data to the Hanford Laboratories Operation.

2. Providing disposal facilities,* and securing specifications and approvals for disposal.

3. Properly disposing of the waste, assuring that the disposed waste is within approved specifications.

4. Sampling and analyzing the waste.

5. Appropriately reporting, demonstrating and documenting the reliability of its inventory data.

6. Reporting each instance of shoreline discharge of reactor effluent or similar waste to the Regional Monitoring Operation by the next workday. Reporting unusual discharges of any waste as specified in Radiation Protection Standard 6.1.

The originating component's responsibility for the waste shall end when the material has been released to the environment in the approved fashion. The point of release is defined to be the end of the discharge pipe or spillway in the river, or the bottom of the ground disposal facility.

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* Refer to HAPO Organization and Policy Guide 8.1 and 8.2 for responsibilities of landlord and using department that apply to waste disposal.

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RESPONSIBILITIES OF HANFORD LABORATORIES OPERATION

The following responsibilities are to be appropriately assumed by functional organizations within Hanford Laboratories Operation.

1. Evaluation of radioactive liquid wastes in relation to associated disposal facilities, employing the best information available from Hanford Laboratories Operation and other sources.

2. Advising the component originating the waste on limits to volume, concentration or total amounts, and on location of disposal facilities to take best advantage of river flows, soil depths, ground water movement or existing facilities.

3. Specification of radioisotopes and other waste characteristics for which analyses are required, and the required analytical accuracy.

4. Notification when discharge to any particular ground disposal facility shall cease.

5. Advising the concerned "user" component on current release to the Columbia River of reactor effluent which fails to meet established disposal specifications.

6. Conducting periodic audits of liquid waste sampling, analyzing methods, and calculation of inventory data.

7. Stating the location, size, depth and special requirements of wells necessary for monitoring purposes; performing such inspection and acceptance tests for the Hanford Laboratories Operation as may be necessary.

8. Obtaining well drilling logs, samples, geological, hydrological and radiological data during drilling operations.

9. Conducting such sampling, testing and probing of the wells as may be necessary to maintain data on the status of the disposal facilities and the fate of the liquid wastes discharged to the ground.

10. Plant assistance type studies and experiments to establish basic disposal criteria.

11. Maintain current chronological and cumulative waste records.

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GENERAL ELECTRIC

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HANFORD ATOMIC PRODUCTS OPERATION

RADIATION PROTECTION STANDARDS

ISSUED BY THE MANAGER RADIATION PROTECTION	SUBJECT: MAXIMUM PERMISSIBLE CONCENTRATION IN AIR, WATER AND THE HUMAN BODY	CLASSIFICATION: Permissible Limits

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
1	H^3 (tritium, T, as HTO or T_2O)	Body	1000	j, b	0.03	j, b	2×10^{-6}	j, b
		GI			0.2	a	6×10^{-6}	a
4	Be^7	Bone	725	a	1	a	5×10^{-6}	a
		GI			2×10^{-2}	a	3×10^{-6}	a
6	C^{14} (as CO_2)	Fat	260	a	3×10^{-3}	a	1×10^{-5}	a
		GI			6×10^{-2}	a	1×10^{-5}	a
9	F^{18}	Bone	5	a	0.2	a	3×10^{-5}	a
		GI			>0.2	a	$>3 \times 10^{-5}$	a
11	Na^{22}	Body			1×10^{-3}	c, b		
		GI			3×10^{-3}	e	6×10^{-7}	e
	Na^{24}	Body	5	a,b	3×10^{-3}	a, b	7×10^{-7}	a, b
		GI			8×10^{-3}	a	1×10^{-6}	a
		Immersion			2×10^{-4}	c		
		Surface			4×10^{-4}	c		
14	Si^{31}	Body	90	f	0.1	d	3.5×10^{-5}	f
		GI			9×10^{-3}	e	1.5×10^{-5}	e
		Other Organs			0.3	d		
		Immersion			3×10^{-3}	c		
15	P^{32}	Bone	10	a	2×10^{-4}	a	1×10^{-7}	a
		GI			8×10^{-4}	a	1×10^{-7}	a
		Immersion			3×10^{-3}	c		
	P^{33}	Bone			8×10^{-4}	c, a		
16	S^{35}	Skin	300	a	5×10^{-3}	a	10^{-6}	a
		GI			6×10^{-3}	a	10^{-6}	a
17	Cl^{36}	Body	80	a, b	1×10^{-3}	a, b	2×10^{-7}	a b
		GI			10^{-2}	a	2×10^{-6}	a
		Other			4×10^{-3}	a	6×10^{-7}	a
	Cl^{38}	GI			4×10^{-3}	e	7×10^{-7}	e
18	A^{41}	Body	11	a, b	2×10^{-4}	a, b	2×10^{-7}	a, b
19	K^{42}	Muscle	21	a	10^{-2}	a	2×10^{-6}	a
		GI			3×10^{-3}	a	6×10^{-7}	a

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden µc		Maximum Permissible Concentration in: Water µc/cc		Air µc/cc	
20	Ca^{45}	Bone	14	a	10^{-4}	a	8×10^{-9}	a
		GI			2×10^{-2}	a	3×10^{-6}	a
	Ca^{47} (+ Sc^{47})	Bone	1	d	4×10^{-4}	d	2×10^{-8}	d
		GI			5×10^{-4}	e	1×10^{-7}	e
21	Sc^{44}	Spleen			1	c		
	Sc^{46} (85 d)	Liver	5	a	0.3	a	5×10^{-8}	a
		Spleen			0.4	a	7×10^{-8}	a
		GI			4×10^{-4}	a	7×10^{-8}	a
	Sc^{47}	Liver	11	a	3	a	6×10^{-7}	a
		Spleen			4	a	9×10^{-7}	a
		GI			9×10^{-4}	a	2×10^{-7}	a
	Sc^{48}	Liver	3	a	1	a	3×10^{-7}	a
		Spleen			3	a	6×10^{-7}	a
		GI			4×10^{-4}	a	7×10^{-8}	a
23	V^{48}	Bone	10	a	0.3	a	6×10^{-7}	a
		GI			3×10^{-4}	a	5×10^{-8}	a
24	Cr^{51}	Kidneys	600	a	0.7	a	10^{-5}	a
		GI			0.02	a	4×10^{-6}	a
25	Mn^{52}	GI			3×10^{-4}	e	6×10^{-8}	e
	Mn^{54}	GI			1×10^{-3}	e	2×10^{-7}	e
	Mn^{56}	Liver	8	a	0.4	a	4×10^{-6}	a
		Kidneys			0.15	a	4×10^{-6}	a
		GI			3×10^{-3}	a	5×10^{-7}	a
		Immersion			3×10^{-4}	c		
		Surface			1×10^{-3}	c		
26	Fe^{55}	Blood	1000	a	5×10^{-3}	a	7×10^{-7}	a
		GI			0.1	a	2×10^{-5}	a
	Fe^{59}	Blood	13	a	10^{-4}	a	2×10^{-8}	a
		GI			3×10^{-3}	a	5×10^{-7}	a
27	Co^{57}	GI			7×10^{-3}	e	1×10^{-6}	e
	Co^{58} (+ Co^{58}) (9 h)	Liver						
		GI			3×10^{-2}	e	6×10^{-6}	e
	Co^{58} (71 d)	Liver			5×10^{-2}	c		
		GI			1.5×10^{-3}	e	3×10^{-7}	e
	Co^{60}	Liver	3	a	2×10^{-2}	a	10^{-6}	a
		GI			4×10^{-4}	a	8×10^{-8}	a

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BM-7000-056 (5-57) AEC-GE RICHLAND, WASH.

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
27	Co^{61}	Liver			3	c		
28	Ni^{57}	Liver			1	c		
	Ni^{59}	Liver	42	a	0.3	a	2×10^{-5}	a
		GI			4×10^{-3}	a	7×10^{-7}	a
	Ni^{63}	Liver			7	c		
		GI			1×10^{-2}	e	2×10^{-6}	e
	Ni^{65}	Liver			1	c		
		GI			1×10^{-3}	e	3×10^{-7}	e
29	Cu^{64}	Liver	120	a	6×10^{-2}	a	5×10^{-6}	a
		GI			5×10^{-3}	a	9×10^{-7}	a
		Immersion			2×10^{-3}	c		
		Surface			5×10^{-3}	c		
	Cu^{67}	Liver			0.1	c		
30	Zn^{65}	Bone	400	a	6×10^{-2}	a	2×10^{-6}	a
		GI			2×10^{-3}	a	4×10^{-7}	a
	Zn^{69} (+ Zn^{69}) (14 h)	GI			7×10^{-4}	e	1×10^{-7}	e
	Zn^{69} (52 m)	Bone			0.08	c, a		
		GI			2×10^{-2}	e	3×10^{-6}	e
	Zn^{72}	Bone			0.1	c, a		
31	Ga^{72}	Bone	3	a	3	a	10^{-6}	a
		GI			5×10^{-4}	a	10^{-7}	a
	Ga^{73}	Bone			8	g, a		
32	Ge^{69}	Bone			0.06	c, a		
	Ge^{71}	Kidneys	72	a	10	a	4×10^{-5}	a
		GI			0.02	a	3×10^{-6}	a
	Ge^{75}	Kidneys			10	g		
	Ge^{77}	Kidneys			0.5	c		
	Ge^{78}	Kidneys			20	g		
33	As^{73} (+ Ge^{73})	GI			5×10^{-3}	e	8×10^{-7}	e
	As^{74}	Kidneys			4×10^{-2}	c		
		GI			5×10^{-4}	e	9×10^{-8}	e

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
33	As^{76}	Kidneys	11	a	0.2	a	2×10^{-6}	a
		GI			2×10^{-4}	a	4×10^{-8}	a
		Immersion			8×10^{-4}	c		
		Surface			3×10^{-3}	c		
	As^{77}	Kidneys			0.8	c		
		GI			8×10^{-4}	e	1×10^{-7}	e
	As^{78}	Kidneys			5	g		
34	Se^{75}	Bone			0.1	c, a		
		GI			0.025	e	4×10^{-6}	e
35	Br^{82}	Body			2×10^{-3}	c, b		
		GI			3×10^{-3}	e	5×10^{-7}	e
	Br^{83}	Body			0.07	g, b		
37	Rb^{84}	Muscle			3	c		
	Rb^{86} (18.6 d)	Muscle	64	a	3×10^{-3}	a	4×10^{-7}	a
		GI			3×10^{-3}	a	4×10^{-7}	a
	Rb^{87}	GI			3×10^{-2}	e	6×10^{-6}	e
38	Sr^{87}	Bone			0.1	c, a		
	Sr^{89} (54 d)	Bone	2	a	7×10^{-5}	a	2×10^{-8}	a
		GI			7×10^{-4}	a	10^{-7}	a
	Sr^{90} (+ Y^{90})	Bone	1	a	8×10^{-7}	a	2×10^{-10}	a
		GI			10^{-3}	a	2×10^{-7}	a
	Sr^{91} (+ Y^{91} + Y^{91}) (50 m)	Bone			8×10^{-4}	g, a		
		GI			6×10^{-4}	e	1×10^{-7}	e
	Sr^{92} (+ Y^{92})	Bone			2×10^{-3}	c, a		
		GI			7×10^{-4}	e	1×10^{-7}	e
39	Y^{88}	Bone			0.004	c, a		
	Y^{90}	Bone			0.6	g		
		GI			2×10^{-4}	e	3×10^{-8}	e
	Y^{91} (+ Y^{91}) (50 m)	GI			3×10^{-2}	e	6×10^{-6}	e
	Y^{91} (58 d)	Bone	3	a	4×10^{-2}	a	9×10^{-9}	a
		GI			3×10^{-4}	a	5×10^{-8}	a
	Y^{92}	Bone			6	g		
		GI			6×10^{-4}	e	1×10^{-7}	e

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BM-7600-036 (8-57) AEC-GE RICHLAND, WASH.

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
39	Y^{93} (+ Zr^{93} + Nb^{93})	Bone			2	g		
		GI			3×10^{-4}	e	5×10^{-8}	e
40	Zr^{93} (+ Nb^{93})	Bone						
		GI			8×10^{-3}	e	1×10^{-6}	e
	Zr^{95} (+ Nb^{95})	Bone	10	a	0.4	a	8×10^{-8}	a
		GI			6×10^{-4}	a	10^{-7}	a
	Zr^{97} (+ Nb^{97} + Nb^{97}) (1 m)	Bone			1	g, a		
		GI			2×10^{-4}	e	3×10^{-8}	e
41	Nb^{92}	Bone			2×10^{-3}	c, a		
	Nb^{93}	Bone			2×10^{-3}	c, a		
		GI			5×10^{-3}	e	9×10^{-7}	e
	Nb^{95} (35 d)	Bone	44	a	2×10^{-3}	a	2×10^{-7}	a
		GI			2×10^{-3}	a	3×10^{-7}	a
	Nb^{97}	Bone			1×10^{-2}	c, a		
		GI			9×10^{-3}	e	2×10^{-6}	e
42	Mo^{99}	Bone	17	a	5	a	6×10^{-4}	a
		GI			3×10^{-3}	a	5×10^{-7}	a
43	Tc^{96} (4.3 d)	Kidneys	5	a	3×10^{-2}	a	3×10^{-6}	a
		GI			10^{-3}	a	2×10^{-7}	a
	Tc^{97} (+ Tc^{97}) (91 d)	GI			4×10^{-3}	e	6×10^{-7}	e
	Tc^{97} (10^5 y)	GI			2×10^{-2}	e	3×10^{-6}	e
	Tc^{99} (+ Tc^{99}) (6 h)	Kidneys			5	c		
		GI			6×10^{-2}	e	1×10^{-5}	e
	Tc^{99} (2×10^5 y)	Kidneys			0.9	c		
		GI			3×10^{-3}	e	5×10^{-7}	e
44	Ru^{97} (+Tc^{97IT} + Tc^{97})	Kidneys						
		GI			4×10^{-3}	e	6×10^{-7}	e
	Ru^{103} (+ Rh^{103})	Kidneys						
		GI			9×10^{-4}	e	2×10^{-7}	e
	Ru^{105} (+Rh^{105IT} + Rh^{105})	Kidneys						
		GI			1×10^{-3}	e	2×10^{-7}	e
	Ru^{106} (+ Rh^{106})	Kidneys	3	k	0.004	k	5×10^{-8}	k
		GI			10^{-4}	a	2×10^{-8}	a

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden µc		Maximum Permissible Concentration in: Water µc/cc		Air µc/cc	
45	Rh^{103IT} (54 m)	Kidneys						
		GI			0.2	e	3×10^{-5}	e
	Rh^{105}	Kidneys	9	a	0.4	a	2×10^{-6}	a
		GI			10^{-3}	a	2×10^{-7}	a
46	Pd^{103} (+ Rh^{103})	Kidneys	7	a	10^{-2}	a	8×10^{-7}	a
		GI			5×10^{-3}	a	9×10^{-7}	a
	Pd^{109}(+ Ag^{109})	Kidneys			2×10^{-2}	c		
		GI			9×10^{-4}	e	1.5×10^{-7}	e
	Pd^{112}	Kidneys			3×10^{-3}	c		
47	Ag^{105}	Liver	1[illegible]	a	2	a	10^{-5}	a
		GI			4×10^{-4}	a	7×10^{-8}	a
	Ag^{106}	Liver			2	c		
		Liver			4	c		
	Ag^{110IT}(+Ag^{110})	GI			3×10^{-4}	e	5×10^{-8}	e
	Ag^{111}	Liver	39	a	5	a	3×10^{-5}	a
		GI			5×10^{-4}	a	8×10^{-8}	a
	Ag^{112}	Liver			17	g		
48	Cd^{109} (+ Ag^{109})	Liver	45	a	0.07	a	7×10^{-8}	a
		GI			0.7	a	10^{-4}	a
	Cd^{115} (+ Cd^{115} +In^{115IT}+In^{115}) (43 d)	Liver			6×10^{-3}	c		
		GI			2.5×10^{-4}	e	4×10^{-8}	e
	Cd^{115}(+In^{115IT} +In^{115})(54 h)	Liver			0.15	c		
		GI			3.5×10^{-4}	e	6×10^{-8}	e
	Cd^{117IT}(+Cd^{117})	Liver			0.9	c		
49	In^{114IT}(+In^{114})	GI			2×10^{-4}	e	3×10^{-8}	e
	In^{115IT}(+In^{115})	GI			4×10^{-3}	e	6×10^{-7}	e
	In^{115}	GI			9×10^{-4}	e	2×10^{-7}	e
50	Sn^{113}	Bone	84	a	0.2	a	6×10^{-7}	a
		GI			0.002	a	3×10^{-7}	a
	Sn^{117}	Bone			0.04	c, a		
	Sn^{119}	Bone			0.04	c, a		

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
50	Sn^{121IT} (+Sn^{121})	Bone			2×10^{-2}	c, a		
	Sn^{121} (27 h)	Bone			1	c, a		
	Sn^{123}	Bone			0.006	c, a		
	Sn^{125} (+Sb^{125} + Te^{125})	Bone GI			2×10^{-2} 2×10^{-4}	c, a e		
51	Sb^{122}	Bone GI			1 3×10^{-4}	c, d e	5×10^{-8}	e
	Sb^{124}	Bone GI			0.2 2×10^{-4}	c, d e	4×10^{-8}	e
	Sb^{125}(+ Te^{125})	GI			1×10^{-3}	e	2×10^{-7}	e
	Sb^{126}	Bone			0.4	c, a		
52	Te^{121IT} (+ Te^{121} + I^{121} + Xe^{121})	Kidneys			2×10^{-2}	c		
	Te^{125}	Kidneys GI			8×10^{-2} 2×10^{-3}	c e	3.5×10^{-7}	e
	Te^{127}	Kidneys GI	4	a	3×10^{-2} 7×10^{-4}	a a	10^{-7} 10^{-7}	a a
	Te^{129IT} (+Te^{129})	Kidneys GI	1.4	a	10^{-2} 2×10^{-4}	a a	4×10^{-8} 4×10^{-8}	a a
	Te^{131IT}(Te^{131} + I^{131} + Xe^{131})	Kidneys GI			0.15 6×10^{-4}	c e	1×10^{-7}	e
	Te^{132}	Kidneys			0.4	c		
	Te^{133}	Kidneys			3	c		
53	I^{126}	Thyroid GI			6×10^{-6} 2×10^{-2}	c, a e	3×10^{-6}	e
	I^{130}	Thyroid			2×10^{-4}	c, d		
	I^{131}	Thyroid GI	0.6	a	6×10^{-5} $>6 \times 10^{-5}$	a a	6×10^{-9} $>6 \times 10^{-9}$	a a
	I^{132}	Thyroid GI			8×10^{-4} 4×10^{-3}	c, d a	7×10^{-7}	e
	I^{133} (+ Xe^{133IT} + Xe^{133})	Thyroid GI			1×10^{-4} 6×10^{-3}	c, a e	1×10^{-6}	e

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
53	I^{134}	Thyroid						
		GI			6×10^{-3}	e	1×10^{-6}	e
	I^{135} (+ Xe^{135IT} +Xe^{135}+Cs^{135IT} +Cs^{135})	Thyroid			4×10^{-4}	c, a		
		GI			4×10^{-3}	e	8×10^{-7}	e
54	Xe^{133}	Body	100	a, b	1×10^{-3}	a, b	1×10^{-6}	a, b
	Xe^{135}	Body	33	a, b	3×10^{-4}	a, b	7×10^{-7}	a, b
55	Cs^{131}	Muscle						
		GI			0.2	e	3×10^{-5}	e
	Cs^{132}	Muscle			7×10^{-3}	c		
	Cs^{134IT} (+Cs^{134})	Muscle			1	c		
		GI			0.055	e	1×10^{-5}	e
	Cs^{134} (2.3 y)	Muscle			0.8	c		
		GI			0.005	e	8×10^{-7}	e
	Cs^{135}	Muscle						
		GI			0.05	e	8×10^{-6}	e
	Cs^{136}	Muscle			2×10^{-3}	c		
		GI			0.008	e	1×10^{-6}	e
	Cs^{137} (+ Ba^{137})	Muscle	98	a	2×10^{-3}	a	2×10^{-7}	a
		GI			2×10^{-3}	a	2×10^{-7}	a
56	Ba^{131} (+ Cs^{131})	Bone			2×10^{-3}	c, a		
		GI			2×10^{-3}	e	3×10^{-7}	e
	Ba^{133IT} (+Ba^{133})	Bone			2×10^{-2}	c,d		
	Ba^{133} (8 y)	Bone			4×10^{-4}	c, a		
	Ba^{135}	Bone			2×10^{-3}	c, a		
	Ba^{139}	Bone			0.1	g, a		
	Ba^{140} (+ La^{140})	Bone	1	a	5×10^{-4}	a	2×10^{-8}	a
		GI			3×10^{-4}	a	6×10^{-8}	a
57	La^{140}	Bone	7	a	0.3	a	4×10^{-7}	a
		GI			3×10^{-4}	a	5×10^{-8}	a
	La^{141}	Bone			2	g, a		
	La^{142}	Bone			5	g, a		

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
58	Ce^{139}	Bone			0.02	c, a		
	Ce^{141}	Bone			0.4	d		
	Ce^{143} (+ Pr^{143})	Bone			2	g		
		GI			4×10^{-4}	e	7×10^{-8}	e
	Ce^{144} (+ Pr^{144})	Bone	1	a	8×10^{-3}	a	2×10^{-9}	a
		GI			10^{-4}	a	2×10^{-8}	a
	Ce^{144}	Bone			2	g, a		
		GI			9×10^{-4}	e	2×10^{-7}	e
	Ce^{145}	Bone			24	g, a		
59	Pr^{142}	Bone			0.4	c, a		
		GI			3×10^{-4}	e	5×10^{-8}	e
	Pr^{143}	Bone	6	a	8×10^{-2}	a	2×10^{-7}	a
		GI			5×10^{-4}	a	9×10^{-8}	a
	Pr^{145}	Bone			1	g, a		
60	Nd^{147} (+ Pm^{147} + Sm^{147})	Bone			1	g, a		
		GI			6×10^{-4}	e	1×10^{-7}	e
	Nd^{149} (+ Pm^{149})	Bone			14	g, a		
		GI			3×10^{-3}	e	5×10^{-7}	e
61	Pm^{147}	Bone	25	a	0.2	a	4×10^{-8}	a
		GI			0.002	a	4×10^{-7}	a
	Pm^{148}	Bone			0.4	c, a		
	Pm^{149}	Bone			2	g, a		
		GI			4×10^{-4}	e	8×10^{-8}	e
	Pm^{151}	Bone			2	g, a		
62	Sm^{147}	Bone						
		GI			7×10^{-6}	e	1×10^{-9}	e
	Sm^{151}	Bone	90	a	5×10^{-2}	a	3×10^{-9}	a
		GI			8×10^{-3}	a	10^{-6}	a
	Sm^{153}	Bone			18	g, a		
		GI			8×10^{-4}	e	1×10^{-7}	e
	Sm^{156}	Bone			100	g, a		

BM-7000-057 (6 - 57) ABC-GE RICHLAND, WASH.

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
63	Eu^{152} (9 h)	Bone			9	d		
		GI			6×10^{-4}	e	1×10^{-7}	e
	Eu^{152} (13 y)	Bone			0.02	d		
		GI			8×10^{-4}	e	1×10^{-7}	e
	Eu^{154}	Bone	7	a	10^{-2}	a	2×10^{-9}	a
		GI			4×10^{-4}	a	8×10^{-8}	a
	Eu^{155}	Bone			0.06	g, a		
		GI			2×10^{-3}	e	4×10^{-7}	e
	Eu^{156}	Bone			0.2	g, a		
	Eu^{157}	Bone			6	g, a		
64	Gd^{153} (+Eu^{153})	GI			2×10^{-3}	e	4×10^{-7}	e
	Gd^{159}	GI			8×10^{-4}	e	1×10^{-7}	e
65	Tb^{160} (+ Dy^{160})	GI			4×10^{-4}	e	8×10^{-8}	e
66	Dy^{165}	GI			4×10^{-3}	e	7×10^{-7}	e
	Dy^{166} (+ Ho^{166} + Er^{166})	GI			4×10^{-4}	e	7×10^{-8}	e
67	Ho^{166} (27 h)	Bone	4	a	5	a	8×10^{-7}	a
		GI			5×10^{-4}	a	8×10^{-8}	a
68	Er^{169} (+ Tm^{169})	GI			9×10^{-4}	e	2×10^{-7}	e
	Er^{171} (+Tm^{171}IT +Tm^{171})	GI			1×10^{-3}	e	2×10^{-7}	e
69	Tm^{170}	Bone	4	a	6×10^{-2}	a	10^{-8}	a
		GI			5×10^{-4}	a	8×10^{-8}	a
70	Yb^{169}	Bone			3	c		
	Yb^{175}	Bone						
		GI			1×10^{-3}	e	2×10^{-7}	e
71	Lu^{177}	Bone	18	a	6	a	10^{-6}	a
		GI			10^{-3}	a	2×10^{-7}	a
72	Hf^{181} (+Ta^{181}IT)	GI			7×10^{-4}	e	1×10^{-7}	e
73	Ta^{182} (112 d)	Liver	6	a	0.1	a	2×10^{-8}	a
		GI			5×10^{-4}	a	9×10^{-8}	a
	Ta^{183}	Liver			1	c		

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
74	W^{181}	Bone	24	a	0.1	a	5×10^{-6}	a
		GI			7×10^{-4}	a	10^{-7}	a
	W^{185}	Bone						
		GI			1×10^{-3}	e	2×10^{-7}	e
	W^{187} (+ Re^{187IT} + Re^{187})	Bone			0.1	c, a		
		GI			7×10^{-4}	e	1×10^{-7}	e
75	Re^{183}	Thyroid	37	a	0.09	a	9×10^{-6}	a
		Skin			0.3	a	3×10^{-5}	a
		GI			0.002	a	4×10^{-7}	a
	Re^{184} (50 d)	Thyroid			4×10^{-3}	c		
	Re^{184} (2.2 d)	Thyroid			0.2	c		
	Re^{186} (+ Os^{186})	Thyroid			8×10^{-4}	c		
		GI			9×10^{-4}	e	2×10^{-7}	e
	Re^{187}	Thyroid						
		GI			0.025	e	4×10^{-6}	e
	Re^{188} (+ Os^{188})	Thyroid			1×10^{-2}	c		
		GI			6×10^{-4}	e	1×10^{-7}	e
76	Os^{191IT} (+ Os^{191} + Ir^{191})	GI			3×10^{-2}	e	5×10^{-6}	e
	Os^{191} (+ Ir^{191})	GI			2×10^{-3}	e	4×10^{-7}	e
	Os^{193}	GI			6×10^{-4}	e	1×10^{-7}	e
77	Ir^{190} (11 d)	Spleen	21	a	0.2	a	10^{-6}	a
		Kidneys			0.01	a	8×10^{-7}	a
		GI			0.003	a	6×10^{-7}	a
	Ir^{192}	Kidneys	3	a	9×10^{-4}	a	5×10^{-8}	a
		Spleen	3	a	6×10^{-3}	a	3×10^{-8}	a
		GI			5×10^{-4}	a	9×10^{-8}	a
	Ir^{194}	GI			3×10^{-4}	e	6×10^{-8}	e
	Ir^{195}	Kidneys			0.09	c		
78	Pt^{191}	Kidneys	2	a	6×10^{-3}	a	2×10^{-7}	a
		GI			7×10^{-4}	a	1×10^{-7}	a
	Pt^{193}	Kidneys	3	a	5×10^{-3}	a	2×10^{-7}	a
		GI			9×10^{-4}	a	2×10^{-7}	a

Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
78	Pt^{195}	Kidneys			1×10^{-2}	c		
	Pt^{197IT} ($+Pt^{197}$)	Kidneys						
		GI			1×10^{-2}	e	2×10^{-6}	e
	Pt^{197}	Kidneys			4×10^{-2}	c		
		GI			1×10^{-3}	e	2×10^{-7}	e
79	Au^{196}	Liver	8	a	5×10^{-2}	a	2×10^{-7}	a
		Kidneys			5×10^{-3}	a	2×10^{-7}	a
		GI			2×10^{-3}	a	4×10^{-7}	a
	Au^{198}	Liver	3	a	4×10^{-2}	a	2×10^{-7}	a
		Kidneys			3×10^{-3}	a	1×10^{-7}	a
		GI			6×10^{-4}	a	1×10^{-7}	a
	Au^{199}	Liver	9	a	9×10^{-2}	a	4×10^{-7}	a
		Kidneys			8×10^{-3}	a	3×10^{-7}	a
		GI			2×10^{-3}	a	3×10^{-7}	a
80	Hg^{197IT} (Au^{197IT} + Hg^{197})	GI			7×10^{-3}	e	1×10^{-6}	e
	Hg^{197}	GI			2×10^{-2}	e	4×10^{-6}	e
	Hg^{203}	GI			4×10^{-3}	e	8×10^{-7}	e
81	Tl^{200}	Muscle	40	a	2×10^{-2}	a	2×10^{-6}	a
		GI			10^{-3}	a	2×10^{-7}	a
	Tl^{201}	Muscle	310	a	8×10^{-2}	a	7×10^{-6}	a
		GI			9×10^{-3}	a	2×10^{-6}	a
	Tl^{202}	Muscle	230	a	2×10^{-2}	a	2×10^{-6}	a
		GI			5×10^{-3}	a	9×10^{-7}	a
	Tl^{204}	Muscle	200	a	8×10^{-3}	a	8×10^{-7}	a
		GI			1×10^{-3}	a	2×10^{-7}	a
82	Pb^{203} (52 h)	Bone	61	a	0.1	a	7×10^{-6}	a
		GI			2×10^{-3}	a	4×10^{-7}	a
	Pb^{210}(+ dr)	Bone	0.2	a	2×10^{-6}	a	8×10^{-11}	a
		GI			3×10^{-3}	a	4×10^{-7}	a
	Pb^{212} ($+Bi^{212}$ $+Po^{212}$ $+Tl^{208}$)	GI			4×10^{-6}	e	7.5×10^{-10}	e
83	Bi^{210} ($+Po^{210}$)	GI			2×10^{-4}	e	4×10^{-8}	e
	Bi^{212} (+ Po^{212} + Tl^{208})	GI			8×10^{-5}	e	1×10^{-8}	e

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in Water μc/cc		Maximum Permissible Concentration in Air μc/cc	
84	Po^{210} (soluble)	Spleen	0.04	a	3×10^{-5}	a	5×10^{-10}	a
		GI			3×10^{-6}	a	5×10^{-10}	a
	Po^{210} (insoluble)	Lungs	0.02	a			1×10^{-10}	a
85	At^{211}	Thyroid	10^{-3}	a	3×10^{-6}	a	5×10^{-10}	a
		GI			$>3 \times 10^{-6}$	a	$>5 \times 10^{-10}$	a
86	$Rn(Th)^{220}$ measured with daughters	Lungs					10^{-7}	a
	Rn^{222} measured with daughters	Lungs					10^{-7}	a
88	Ra^{224} (soluble)	Bone	0.06	h	5×10^{-5}	h	8×10^{-9}	h
	Ra^{224} (insoluble)	Lungs	0.003	h			3×10^{-10}	h
	Ra^{226} (+55% dr)	Bone	0.1	a	4×10^{-8}	a	8×10^{-12}	a
	Ra^{228} (soluble)	Bone	0.04	h	8×10^{-8}	h	1×10^{-11}	h
	Ra^{228} (insoluble)	Lungs	0.002	h			2×10^{-12}	h
89	Ac^{227} (+ dr)	Bone	0.01	a	3×10^{-6}	a	4×10^{-12}	a
		GI			6×10^{-5}	a	1×10^{-8}	a
90	Th^{232} (natural) (soluble)	Bone	0.01	a	5×10^{-7}	a	3×10^{-13}	h
		GI			1×10^{-6}	a		
	Th^{232} (natural) (insoluble)	Lungs	2×10^{-3}	a			3×10^{-13}	h
	Th^{230}	GI			3×10^{-6}	e	6×10^{-10}	e
	Th^{231}	GI			2×10^{-3}	e	4×10^{-7}	e
	Th^{234} ($+Pa^{234}$)	Bone	2	a	5×10^{-2}	a	1×10^{-8}	a
		GI			2×10^{-4}	a	3×10^{-8}	a
91	Pa^{233}	GI			1×10^{-3}	e	2×10^{-7}	e
92	U^{233} (soluble)	Bone	0.04	a	1.5×10^{-4}	a	3×10^{-11}	a
		GI			3×10^{-6}	a	5×10^{-10}	a
	U^{233} (insoluble)	Lungs	0.016	a			3×10^{-11}	a
	U^{234}	GI			3×10^{-6}	e	5×10^{-10}	e
	U^{235} ($+ Th^{231}$)	GI			3×10^{-6}	e	6×10^{-10}	e

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Z	Radioisotope	Critical Organ	Maximum Permissible Total Body Burden μc		Maximum Permissible Concentration in: Water μc/cc		Air μc/cc	
92	U^{238} (natural) (soluble)	Kidneys GI	0.04	a	1×10^{-4} 2×10^{-6}	a a	3×10^{-11} 3×10^{-10}	a a
	$U^{(238)}$ (natural) (insoluble)	Lungs	0.01	a			3×10^{-11}	a
93	Np^{237} (+ Pa^{233})	Bone GI			4×10^{-6}	e	6×10^{-10}	e
	Np^{239} (+ Pu^{239})	Bone GI	10	i	1 2×10^{-3}	i i	4×10^{-7}	i, e
94	Pu^{239} (soluble)	Bone GI	0.04	a	6×10^{-6} 3×10^{-6}	a a	2×10^{-12} 5×10^{-10}	a a
	Pu^{239} (insoluble)	Lungs	0.02	a			2×10^{-12}	a
95	Am^{241}	Bone GI	0.06	a	2×10^{-4} 3×10^{-6}	a a	4×10^{-11} 5×10^{-10}	a a
96	Cm^{242}	Bone GI	0.06	a	1×10^{-3} 2×10^{-6}	a a	2×10^{-10} 4×10^{-10}	a a

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APPENDIX A

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84—7000—002

HANFORD LABORATORIES OPERATION
GENERAL ELECTRIC
RICHLAND, WASHINGTON

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April 4, 1955

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CURRENT VIEWS ON THE I^{131} EMISSION LIMITS

The official limit for permissible concentration of radioiodine in the atmosphere is that given in NBS Handbook 52. This relates to the breathing hazard. Since the first year of Hanford operation, we have known that this limit is too high because the real hazard arises from deposition of the active material on vegetation. The limiting hazard is either

Case I ingestion by range animals or
Case II ingestion by humans.

Case I

Our experimental work has so far been restricted to sheep. We now find deleterious affects at 5 μc I^{131} per day. The probable safe limit is ~ 1 μc I^{131} per day. For other reasons, we use a vegetation limit of 10^{-5} μc per gm. This gives 8×10^{-2} μc per day to sheep, which should have a safety factor of about 10.

However, at this level, the thyroid dose is quite high:

Daily intake = 8×10^{-2} μc

Permanent I^{131} burden = ~4 x daily intake
= 0.32 μc

Thyroid dose rate = $\frac{360 \times E \times Q}{W}$ rads per week

Where E = effective energy in MEV = 0.22
Q = permanent burden (μc) = 0.32
W = mass of thyroid gland (gm) = ~8
Dose rate = ~3.2 rads per week

For cattle, the probable dose rate is ~7 rads per week, for the same contamination.

Case II

The hazard is a combination of

(1) direct inhalation - usually negligible
(2) eating of fresh garden produce
(3) drinking of milk from cows on contaminated pasture
(4) drinking of contaminated water - usually negligible.

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With the provisional limit of 10^{-5} μc per gm, we have:

Permissible daily intake for general public = ~0.006 μc I^{131}.

Permissible intake of cow's milk = 2 quarts per day.

Permissible intake of garden produce = ~600 gm per day.
Or appropriate combinations from milk and food.

For children, the values should probably be less.

These numbers contain no safety factors beyond those inherent in national limits, and the quantities involved are within feasible limits on some farms. It follows that for Case I and Case II, there is no incentive to increase the vegetation limit beyond 10^{-5} μc per gram. There is a possibility that improved analyses of the "home economics" of the human case will drive the value down.

In addition, to operate at this limit would require regional radiation monitoring in such detail as to answer the possible questions for any farm in the affected area. It would be far wiser to stay well below the limit at all times.

Conversion to stack emission limits

The remaining problem is to establish a correlation between ground deposition and stack emission. This can be done in two ways:

(a) use standard equations such as those of Sutton to determine concentration of I^{131} in air brought to the ground Determine a coefficient that measures the vegetation contamination in a given I^{131} concentration in the atmosphere, or

(b) by experience, determine the vegetation contamination for known stack emission.

Neither method is good to a factor of 2. Our present limit of one curie I^{131} per day is a good working compromise of methods (a) and (b).

It was developed when only one stack was operative, and we have given no clear definition as to whether we meant one curie per day for Hanford or for each plant.

The true control limit is the vegetation contamination. We therefore agreed to consider one curie per day per stack as a reasonable tentative target. The actual emission limit will be a function of the meteorological parameters, and will vary with the seasons. To oversimplify, the winter season, with preponderance of stable air, means safe operation near the plants but widespread contamination in the environs; summer brings downdraft conditions near the plant, with nuisance contamination, and generally good dilution at more remote points.

The impossibility of giving a firm limit for daily emission short of setting one so low as to be absurdly restrictive, is obvious. A review of the actual vegetation contamination suggests that a Hanford total of one curie I^{131} per day would always be safe. If this target can be assured by reasonable expenditures, it is recommended. Periods of a maintained average of 2 curies per day have produced off-project contamination above limits.

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RICHLAND, WASHINGTON

J. E. Maider -3- April 4, 1955

As a probably effective alternate, we suggest that the working limit (Hanford total) be set at 10 curies per week, with not more than 3 curies on any one day. In both cases, it is understood that dissolving will not take place when the Synoptic Meteorology Sub-Unit indicates adverse dilution conditions.

To work to higher limits would require stringent control of dissolving times at some seasons, which is not recommended. We have almost completed development of a substantially improved meteorological control system that should improve acceptance of controls necessary in the recommended limits.

Summary

1. After a period of suggesting more relaxed limits for I^{131} emission, the trend of present studies both in animals and in man, is in the reverse direction.

2. Measurement of adequate performance in this field is in terms of vegetation contamination with a limit of 10^{-5} μc per gram.

3. Operating control is by observance of a daily or weekly emission limit, which cannot be rigorously defined, and by avoidance of dissolving during times of poor atmospheric dilution.

4. A Hanford total of one curie I^{131} per day is proposed as the desirable maximum.

5. A more relaxed limit, which would probably be satisfactory, is 10 curies per week with not more than 3 curies in any one day.

   It is anticipated that this limit would not lead to serious interference with production as a result of control to meteorological requirements.

6. Higher limits could be used, provided considerable control to meteorological requirements is accepted. Such a course is not recommended.

ORIGINAL SIGNED BY
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THE STORAGE OF HIGH LEVEL RADIOACTIVE WASTES DESIGN AND OPERATING EXPERIENCE IN THE UNITED STATES

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ORIGIN OF WASTES

In the course of operating nuclear reactors for power generation or for the production of fissionable material, the reactivity of the fuel decreases because of burnout of the fissionable material and by buildup of interfering concentrations of fission products, some of which have a high capacity for absorbing neutrons. Those isotopes which are formed by fission and which have a high neutron capture cross section are called poisons. Because of burnout of the fuel and concurrent "growing in" of the poisons, the fuel must be replaced before all of its fissionable material content has been consumed. To assure economical operation of nuclear reactors, the spent fuel must be processed chemically to recover the remaining fissionable (and fertile) materials by methods which separate them from the extraneous and objectionable fission products. The purified fissionable (and fertile) materials obtained by this separations process can then be reconverted to fuel for reuse in reactors. The achievement of economical atomic power depends on the realization of low costs in all stages of the fuel cycle of which the separations process is a part.

In the separations processes the fission products along with the many conventional chemical agents necessary for the operation of the process emerge as wastes. Although ideally a separations process should produce a single waste stream, such is not the case in practice. Effluents such as steam condensates, cooling water, condensed vapors evolved from boiling radioactive process solutions and streams in contact with waste gases may on analysis have sufficient radioactive contamination to require storage or further processing for clean-up and eventual disposal. In view of the great biological hazard associated with the intense radioactivity of a portion of the fission products, these wastes, as determined by the nature and concentration of these fission products, must be securely stored indefinitely. Such storage is the eventual destination of the hazardous waste derived from the fuel cycle of all nuclear reactors. The facilities provided in the United States for the permanent storage of such materials and some of the accumulated

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operating experience with these is the subject of this paper.

NATURE OF THE WASTES

High level wastes may be described and characterized generally as being either acid or alkaline, concentrated, aqueous, salt solutions, hazardous to all living organisms because of intense radioactivity and capable of spontaneous and prolonged boiling because of absorption of their own radiant energy. These waste streams contain the major portion of radioactive fission products and therefore require storage facilities which will provide secure containment for an indefinite period. The hazardous nature of these fission products, although existing in concentrations which in ordinary industrial chemical operations are regarded as fantastically small, require most extreme precautions in their handling and disposition. For example, toxic materials[1] such as chlorine, cyanide, arsenic, lead, and mercury are tolerated in common industrial effluents in concentrations amounting to a few parts per ten million (10^7). However, the figures for maximum permissible concentrations (MPC) for certain radioactive fission products[2], if expressed on the same basis, are smaller by a factor of one hundred million (10^8) and would be expressed on a weight basis as parts per 10^{15} parts of effluent. With such requirements the disposal of a number of effluents in the nuclear processing industry by the usual industrial methods is completely intolerable. At present earth covered metal and concrete tanks provide an economical and reliable method for the storage of high level radioactive wastes. The utmost in corrosion resistance of the container material is required. Since these containers are buried underground, they must be built with adequate strength to resist high external earth pressure and also in certain cases to resist the pressure of the ground water. In addition, the property of the waste to heat themselves and boil irregularly results in pressure changes and thermal variations which require special design and structural considerations.

Essentially all wastes being generated on a large scale today are derived from solvent extraction-type processes for uranium and plutonium decontamination and recovery. The combined wastes from the series of extraction cycles comprise the high level wastes of interest here. The wastes which are stored in ordinary steel must be maintained alkaline. Wastes are also stored in the acid condition in stainless steel vessels. In certain operations, the added costs of such stainless steel storage can be justified. However, in the very large scale operations involving the storage of many millions of gallons of waste, concentrated alkaline compositions are retained exclusively in ordinary steel. It has been reported that at the Hanford site alone there are about 50,000,000 gallons of waste stored in such facilities.

COMPOSITION OF WASTES

The approximate composition of typical high level alkaline wastes in major components as delivered to storage is as follows:

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	Molarity	
	Redox	Purex
$Na_2U_2O_7$	Trace	Trace
$NaNO_3$	4.5	4.5
Na_2SO_4	0.05	0.2
Na_2CrO_4	0.3	--
$NaAlO_2$	1.3	Trace
NaOH	1.2	0.2
$Fe(OH)_3$	Trace	0.2
Approximate heat generation rate, watts/liter	0.5	8.0
Specific gravity	1.37	1.15

THE HEAT GENERATING PHENOMENON

When the plutonium or uranium fuel atoms undergo fission, the fragments comprise stable and radioactive isotopes of a number of elements. The radioactive isotopes vary widely in their half-lives. Fortunately many have half-lives so short that by the time the separations process is applied they have been transformed into stable elements. Those with half lives sufficiently long remain in the wastes from the separations processes. Their beta and gamma radiation, which is continuously released, is absorbed in the total stored volume and produces heat. This form of heat generation in a fluid is a unique industrial phenomenon, particularly when it is noted that such heat generation is taking place throughout each increment of the whole volume. Furthermore, such heat generation is not controllable by any known method. The fact that the heat generation rate decreases as the radioactive isotopes decompose to stable elements is not a significant factor in design because the highest initial rates must be provided for.

The decay of the fission products associated with the stored wastes produces heat at a rate determined by the extent and intensity of the irradiation of fuel in the nuclear reactor and by the length of time the fission products have "cooled" since reactor discharge. Typical heat generation rates resulting from fission product decay are shown in Figure 1 for a fuel initially containing 1.2 per cent U-235 and irradiated at five megawatts per ton for 500 days. It will be noted that the overall rate of heat evolution steadily declines, although the rate of decrease slows with time.

The accumulated heat generation associated with a waste storage tank will be somewhat different than the simple case just depicted. The general trends may be illustrated by Figure 2. During the time that the wastes from a separations plant are added to a given tank, the rate of heat generation in that

tank increases, depending on the rate of uranium processing and its irradiation and cooling history. The peak heat generation attained is a function of the same variables. When the addition of fission products to the tank ceases, the heat generation rate is at a maximum and continues to decrease as the fission products decay. During the first year of storage, the rate of heat generation is governed by the decay of fission products with a short half-life (e.g., 60 days or less). During this period the heat generation rate rapidly decreases. Within a few years the rate of heat generation falls to a small fraction of the peak rate; further decreases are gradual as controlled by the longer-lived fission products. In concentrated wastes the heat generated is sufficient to cause the wastes to boil for long periods of time due to the decay of the long-lived fission products (cesium and strontium). For example, operating conditions are possible which would create a waste that would boil for 100 years. To some extent such boiling is utilized to reduce the volumes stored by letting the wastes concentrate themselves.

DESCRIPTION OF HANFORD HIGH LEVEL WASTE STORAGE FACILITIES

High level wastes from fuel processing operations at the Hanford plant are stored in steel-lined, reinforced concrete tanks equipped with external condensers for removal of the heat produced by radioactive decay. A sketch of a typical high level storage system is shown as Figure 3.

Wastes from the separations plants are routed to the storage tank group, or farm, by means of an underground system of stainless steel piping contained in a reinforced concrete trough or encasement. A zig-zag or "dog-leg" layout provides for free expansion and contraction caused by thermal changes. A photograph of the construction of a portion of the encased waste pipe lines is shown in Figure 4.

At each tank farm the waste delivery lines terminate in a diversion box from which the wastes can be directed to the individual tanks. The diversion box is equipped with switching pipes or jumpers by which the routings of a waste stream can be changed or diverted from one tank to any other. The diversion box is constructed of thick concrete for radiation shielding and is connected to a collection tank in which any leakage in the box or attached encasements can accumulate. Such accumulation can also be delivered to the storage tanks. As tanks become filled, routings are changed by mechanically repositioning the jumper connections by remote handling methods. A photograph of such an operation is shown as Figure 5. Operators are able to see their work only by the use of the large mirror placed over the box. Such methods are necessary to avoid exposure to the radiation coming from the waste lines in the diversion box.

This operating technique which is used in most large scale facilities for processing radioactive materials in the United States depends on construction of the diversion box nozzles and fabrication of the jumpers to extremely close tolerances. Figure 6 is a photograph of a typical jumper. An electrically operated impact wrench which hangs from a crane drives the single nut on each connector head to connect and disconnect the jumper. The remotely operated impact wrench is the only tool required to conduct the operation of diverting the delivery of waste from one tank to another. Experimental installations of standard industrial-type valves which were operated with long handles

extending through the thick concrete shielding have been made. These test installations have not successfully demonstrated advantages over the jumper method of remote operation.

The tanks are usually constructed in groups of multiple units, each with a capacity from 500,000 to 1,000,000 gallons, with an internal diameter of approximately 75 feet, and a height from 20 to 40 feet. A view of a tank farm during construction is shown in Figure 7. The steel liner covers the bottom and sides of the reinforced concrete tank. The tank fill lines enter near the top of the steel liner. Overflow lines may or may not be provided, depending upon whether or not it is desired to allow the liquids to cascade from one tank to another.

A number of nozzles enter the tank through the dome to provide means for temperature, pressure, liquid level, and radiation measurement, and for sampling and installation of auxiliary equipment for liquid transfer or agitation. Figure 8 shows the cross section of a typical Hanford waste storage tank. Other equipment shown in Figure 3 is installed to control the problems associated with the removal of the heat generated by radioactive decay of the fission products being stored. A seal pot is provided for protection against excessive pressure changes in the waste storage tank system. The vapors from the storage tanks pass through a device for removal of entrained liquids and are then condensed. Uncondensed and non-condensible vapors pass through a filter and fan to a stack for atmospheric dispersal. Liquid effluents from the filter and condenser proceed to a ground disposal basin or are returned to the storage tanks. Wells are drilled in the storage area to allow the earth to be monitored for evidence of tank leakage and performance of other neighboring waste disposal facilities.

The capital investment and operating cost associated with an underground waste tank storage facility are greatly affected by the size of the facility and by the nature of the surroundings. However, facilities with several tanks having a 500,000 to 1,000,000 gallon capacity can be constructed in dry, easily excavated earth for an overall cost of about 25 to 30 cents per gallon. With elaborate vapor handling systems and thicker steel for increased corrosion resistance for self-boiling wastes, the costs run up to about 40 cents per gallon.

INSTRUMENTATION AND CONTROL OF THE HANFORD WASTE STORAGE SYSTEM

The problems associated with operating a boiling, high level, radioactive waste, storage facility include: a) monitoring the material in each of the tanks to note liquid levels, sludge levels, and temperatures, b) assuring adequate heat removal, and c) eliminating or minimizing the surge boiling characteristics of such wastes. Thus, each waste storage tank is provided with the folowing accessories for measurement and control:

a) A pressure indicator and alarm.

b) A weight factor indicator and alarm (for liquid level).

c) A specific gravity indicator.

d) A conductivity-type level indicator (to check liquid levels).

e) A T-type hand-operated probe (to determine sludge levels).

f) Thermal elements (for temperature measurement).

g) Air lift circulator differential pressure recorder.

h) Air lift circulator air supply flow indicators.

i) Alarm system for the two compressors supplying motive air to the circulators.

In addition, each group of tanks has liquid level and alarm instrumentation to monitor the loop seals shown in Figure 3. The liquid and sludge stored in the tanks are sampled on a non-routine basis by lowering suitable devices through the access manholes and/or spare risers on each tank.

OPERATING EXPERIENCE WITH HANFORD WASTES

A property of the heating phenomenon which is especially unusual in the large volumes is the characteristic of irregular rate of heat release. Wastes may be releasing heat (boiling) at a moderate rate and then periodically burst into a violent surging boil. At such periods the heat release rate may be more than twenty times the normal uniform rate. Under these conditions pressurization of the tanks occurs. Such events are felt to be caused by the release of heat stored under uniform conditions of hydrostatic head throughout the large volume rather than by superheating phenomena. Although heat may be released from a tank by uniform boiling at rates of as much as a few million BTU per hour, this is indeed a slow rate of heat evolution from the liquid surface in a tank which may be seventy-five feet in diameter. Turbulence would not be expected under such conditions which may release heat uniformily at rates even as high as 1000 BTU per hour per square foot. Bubble formation may not even be attained. With the whole volume at equilibrium it can be seen that much heat could be retained without boiling, particularly in the deep part of the tank volume where the temperature would be a few degrees above the atmospheric boiling temperature. Something is assumed to upset the equilibrium and initiate the peculiar surging boil. Alkaline wastes contain precipitates which effectively carry fission products. Such precipitates settle and produce a layer of much higher heat generation capacity at the bottom of the tanks. It may be assumed that the equilibrium is upset by bubble formation at high temperature areas probably in the sludge layer. On escaping, this bubble expands and draws into its upward path liquid which bursts into a boil as its vapor pressure exceeds the hydrostatic head of its immediate environment. Thus, a surge of boiling is propagated and continues until a large fraction of the stored heat is liberated. However, even during the course of this event the surge boiling increases and decreases in intensity in a fairly regular manner. Cyclic pressurization has thus been observed. This is believed to be the self dampening effect on boiling caused by back pressure in restrictions in the vapor release facilities. It is concluded that the maximum pressure attainable under surge boiling cannot exceed the maximum hydrostatic head of the liquid in the tank. This

limit is used as a basis for the design of new facilities. It is also the basis for storage tank operation, in that facilities are provided which assure that such pressures will not be exceeded. The maximum pressure observed during surge boiling has been about two pounds per square inch which is substantially below the maximum hydrostatic head.

With the development of the solvent extraction processes with their minimum waste volumes as distinguished from the original precipitation method, the heat generation phenomenon reached levels which required the provision of elaborate cooling facilities for all tanks containing high level wastes. During the operation of the obsolete precipitation process for plutonium production, self heating of the wastes was expected. Simple finned-tube air-cooled condensers were adequate for condensing the water vapor from the small amount of vent gases. These wastes, which were of large volume since no effort was made to concentrate them for minimum storage costs, did not attain boiling temperatures. Solvent extraction process wastes, however, because of intensive efforts to minimize salt content and to reduce volume, rapidly attain boiling conditions and are expected to continue to boil for many years. (Figure 12)

In order to eliminate cyclic, uncontrolled boiling, the concept of an air lift agitator or circulator for the liquid was examined, demonstrated in large-scale facilities and subsequently installed in all high level waste storage tanks. Figures 9 and 10 show demonstrations of circulators being tested in cold fresh water.(3) Although such equipment is simple and highly satisfactory for eliminating uncontrolled surge boiling, extensive auxiliary facilities must be provided to assure prolonged uninterrupted operation. For example, in highly concentrated wastes the air flow to the circulators is impeded and eventually interrupted by solid formation, crystallization, at the air inlet. Airflow to each circulator is therefore continuously charted. As the rate falls, alarms are actuated. Water is then manually routed to the air line to dissolve the interfering material and restore the flow of air. Figure 11 illustrates the application of the air lift circulators; two pairs of different lengths are installed to permit agitation at low and high liquid levels as the tank is filled.

Temperature profile measurements in the tanks have shown no unexplainable abnormalities. The normal gradients due to hydrostatic head phenomena are found,and higher temperatures are found at the bottom in the sludges. In general, these latter temperatures are not more than 70 C higher than associated liquid temperatures. However, sludge temperatures have in some cases been observed to be 100 C above the liquid temperatures.

Experimental evidence also indicates that hydrogen and oxygen are evolved by radiolysis of stored water and salts. Repeated samplings have never detected hazardous concentrations of hydrogen in the Hanford tanks. It is postulated that the hydrogen is purged from boiling tanks by the flow of steam and from non-boiling tanks by the natural circulation of air through the vapor spaces of the vessel. This problem is receiving continued study.

UNDERGROUND WASTE STORAGE AT THE SAVANNAH RIVER PLANT

The underground waste tank storage facilities at the Savannah River Plant are similar to those at Hanford except for the method of removing the heat

generated by radioactive decay and the specific construction details of the tanks required by the more restrictive environmental conditions (ground water proximity, population density, etc.) at the Savannah River site. The general details of the storage tanks are shown in Figure 13. A "cup and saucer" arrangement is utilized to insure the safe collection of any leakage that might occur. Monitoring equipment is installed in the free space that exists between the outer steel-lined concrete shell and the steel tank. Each tank is approximately 75 feet inside diameter by 25 feet high, with a capacity of 750,000 gallons. The tanks are constructed of one-half inch steel and have a steel cover which together with the dome and earth covering is supported internally by twelve concrete columns. The columns are sheathed in steel which is welded to the top and bottom of the tank. Instrumentation is similar to the Hanford installation.

Heat generated by radioactive decay is removed by means of cooling coils. Thus, the operating temperatures of the tank are held considerably below the boiling point. The bottom cooling coils are circular, steel coils with a surface area of 15,000 square feet to provide cooling at the initial stages of filling. After an appreciable quantity of liquid is collected in the tank, additional cooling is provided by a grid of vertical pipes consisting of a set of 18 to 20 parallel branches on four foot equilateral spacing having about 9000 square feet of heat exchange surface. The vertical coils are more efficient for heat removal. Allowance for additional cooling is provided by another set of vertical coils installed at the top of the tank to act as a spare.

In addition to the customary appurtenances such as catch tanks and pipe tunnels, reflux condensers, entrainment separators, and particulate filters are provided on the waste storage tanks to minimize the chances of discharging vapors contaminated with radioactive materials, if the wastes should boil. Hydrogen accumulation is avoided by injecting an air purge.

The expense of these storage tanks is increased because of the "cup and saucer" design and other construction features required by environmental conditions. Construction costs for a farm with several tanks are estimated at about $1.00 to $1.50 per gallon.

THE IDAHO CHEMICAL PROCESSING PLANT WASTE TANK FARM

For the storage facilities at the Idaho Chemical Processing Plant, process and economic considerations indicated that wastes should be stored under acidic rather than alkaline conditions. Thus, the permanent storage tanks are constructed of stainless steel rather than carbon steel.

The high level wastes are jetted from the process building directly to three cascaded, 18,000 gallon, type 347, stainless steel, temporary storage tanks housed in cells in a waste disposal building. Fission product heat is removed by submerged stainless steel coils and a reflux condenser. When these tanks are filled, the waste is transferred to a permanent waste tank farm consisting of 300,000 gallon, type 347, stainless steel, underground, permanent storage tanks, provided with water cooled reflux condensers, and an instrument house for the remote instrument transmitters on the tank. The tanks are enclosed

in a concrete vault buried under approximately ten feet of earth. Space and connections have been provided for future tanks, when they are required.

The capital investment required to construct such a waste storage facility is estimated at approximately $2.00 to $3.00 per gallon.

DESIGN CONSIDERATIONS FOR HANFORD WASTE STORAGE FACILITIES

The storage facilities must be designed to contain the liquid safely for an indefinite period of time. Provisions must be made for removal and transfer of wastes in case of tank leakage. Adequate shielding is mandatory. The underground location of the tanks aids in this respect, the amount of earth covering being determined by shielding requirements. Therefore, the tanks must be designed to resist this external earth pressure.

The soil encountered at Hanford is firm sand or sand and gravel. A maximum load bearing value which was determined by test to be 8,000 pounds per square foot was utilized in the design. The ground water is far below the bottom of the tanks and can be ignored from a structural standpoint.

Expansion joints in the concrete structure are undesirable because of the possibility of leakage, especially after a period of years. In addition, practical difficulties make subsequent repairs improbable.

The waste solutions are made alkaline to permit the use of carbon steel for the inner liner. The volume of wastes has been such that it has been practical and economical to provide for several million gallons of storage capacity at one time. The specific gravity of the liquid is less when it first enters the tanks, and increases with time as vapors boil away. Self concentration is continued to the point of sodium nitrate crystallization.

To provide for the controlled dissipation of heat formed in the stored solution by the radioactivity of the fission products, finned-tube, air-cooled condensers were mounted on the first underground tanks which were constructed. The small flow of condensate formed was returned to the storage tank. In recent years process improvements provided wastes which permitted self-concentration with resulting economies in storage volume. As the rate of heat generation was increased by these process improvements to reduce waste volumes, air-cooled condensers were superseded by water-cooled condensers of higher capacity. The design of the tanks must take into account the high temperature of the liquid, plus a still higher temperature in the sludge, as well as the effect of temperature differential throughout the structure.

As described earlier, the heat in the form of steam is often released unevenly, in sudden surges, at many times the average heat liberation rate. Since the steam is released much faster than the heat dissipation system can handle it, a buildup of pressure occurs within the tank. The tanks must be strong enough to resist this vapor pressure in addition to the internal liquid pressures, external earth pressure and high temperature. This pressure buildup from heat generation is a rather recent phenomenon occurring under operating conditions different from those for which the storage tanks were originally designed. Hence the existing structures had to be carefully evaluated to determine the amount of internal vapor pressure they would

safely contain. Steps were taken to insure that this pressure was not exceeded.

Finally there are several types of wastes which require separate storage. Separate containers, rather than a single compartmentalized container, were selected since they provide greater safety and flexibility for accommodating process variations and improvements. This flexibility is desirable since the tanks are an integral part of a large and costly separations plant.

EVALUATION OF TYPES

Many types of storage containers have been considered, evaluated, and re-evaluated as the storage conditions have changed. It is believed that the structure described below was the most economical and satisfactory for the specific conditions at the time of construction.

The design selected was a 75 foot diameter, vertical cylinder comprised of a steel tank inside a reinforced concrete tank with an essentially flat bottom of both steel and concrete, and a concrete, ellipsoidal, dome roof.

The underground location aids in shielding from radiation of both tanks and supply and vent lines, and in preventing the rapid spread of any leakage that might occur. The cylindrical shape adapts itself well to changes in temperature. The inner steel liner is more leakproof than would be a tank of reinforced concrete only. The outer concrete tank forms a second line of defense to contain the liquids, if a leak occurs in the steel tank. If both leak, the test wells which are provided in each tank farm aid in discovery after which the tank contents could be transferred to other tanks. The concrete tank resists the external earth pressure and both liquid and vapor pressures.

An ellipsoidal dome was chosen as the shape for the roof because it requires no interior supports. (Because of the lack of ground water, columns are not required at Hanford to resist the uplift from water pressure under the tank.) Its low rise compared to a hemisphere reduces the amount of excavation. It is a good structural shape to support the earth shielding cover, and to resist internal vapor pressure.

The 75 foot diameter was selected because it was as economical as larger diameters, and had the advantage of permitting the bottom to be laid in one pour of concrete without construction joints, thus providing a tighter tank. It was felt that anything over 75 feet should be made in more than one pour.

Other types of tanks were considered and rejected for various reasons. Such considerations are summarized as follows:

A. A single, large, underground, covered, "swimming-pool" type of structure with compartments designed to handle the different types of wastes. The design difficulties were:

   1. Making it adequate to withstand internal pressure.

   2. Assuring an absolutely tight container which would eliminate the possibility of leakage through expansion and construction joints.

3. Providing for the very considerable changes in temperature.

B. A rectangular steel tank with flat top and bottom and closely spaced columns which act as ties to resist internal pressure. This would be protected externally with concrete. The chief problem in this design lay in handling the movement caused by temperature variations.

C. A vertical cylindrical tank with flat top and bottom with columns. This type was rejected for the same reasons as Type B.

D. A vertical cylindrical tank with a flat roof and bottom, the roof being a clear span structure of beam and slab construction. Both ordinary cast concrete and pre-stressed concrete were considered for the roof. Because of the heavy external loads of six to eight feet of earth required for shielding the flat clear span roof was found uneconomical.

E. Steel spheres and spheroids in a concrete box or housing. These were found uneconomical, chiefly because of the underground location. The steel spheroid itself is an economical structure for the combined storage of liquid and gas pressure, but the concrete structure required to protect it from the earth pressure was expensive. The depth required for burial is considerably greater than for other types.

F. Long horizontal cylinders. These were considered and found uneconomical for the relatively low internal pressures involved, chiefly because more material is required per gallon of storage capacity.

G. A group of open-top steel cylindrical tanks in a common concrete housing. This was found uneconomical.

It should be recognized that if design conditions change -- for instance, if vapor pressures increase greatly -- a different type of structure would be more suitable.

OPTIMUM PROPORTIONS OF TANKS

After selecting the type of tank, a study was made to achieve the optimum proportions. The tank referred to here is an underground, cylindrical concrete tank with flat bottom and ellipsoidal dome roof, containing an inner steel liner covering sides and bottoms as shown in Figure 8. A specific gravity of 1.5 was assumed for the liquid waste. In determining the optimum tank dimensions, the quantities of the principal construction materials for various tank sizes were calculated, the cost estimated, and the results evaluated and compared. The chief elements consisted of (1) concrete in place, (2) reinforcing steel in place, (3) form work for the concrete, (4) carbon steel plate liner (inner tank), and (5) excavation and backfill. Accessories such as piping, condensers, test wells, instrumentation, and vapor manifolds were not included since they were assumed to be identical for each type of tank being studied.

The chart presented as Figure 14 shows the economical selection of the tank dimensions. It was found that the cost of storage decreased with an increase in diameter until a 75 foot diameter was reached. For diameters above 75 feet, the cost remained essentially the same up to 130 feet. The optimum

height was different for each diameter. For a tank diameter of 75 feet, the optimum height (depth of liquid) was approximately 35 feet. These results hold good, of course, only for the conditions assumed and the unit prices found to be prevalent at this site and at that time. For instance, if the tanks were placed above ground instead of requiring seven feet of earth cover, different dimensions would probably be more economical.

It was found that the smaller the diameter of the tank, the more critical is the wall height; i.e., a variation in the wall height has more effect on the unit cost. It is to be expected that the deeper the tank the more economical it is up to a certain point because the dome roof remains the same regardless of the tank depth. Also, the deeper tanks require somewhat less excavation and backfill per gallon of capacity.

One of the considerations which limits the depth of the tank is the soil load under the circumferential walls of the tank. The entire earth load above the dome is taken by these walls and the load on the soil beneath must be kept within allowable limits. Differential settlement between the wall footings and the rest of the tank bottom must be kept to a minimum to prevent cracking of the concrete bottom.

Another item which must be factored into selecting the size of the tank is that, if larger and fewer tanks are used, the additional cost of the increased size may be offset by reduced overall costs, since less piping, condensers, and other auxiliary equipment may be required. Based on these and other studies the tank diameter of 75 feet has been utilized at Hanford since the start of the project, but liquid depth has been increased from an original 16 feet to slightly over 30 feet to gain maximum construction economies.

METHOD OF STRUCTURAL DESIGN

Inner Steel Tank

The inner steel tank was designed to the specifications of the American Water Works Association, with certain exceptions. Better welding was required, including complete penetration of all joints, both vertical and horizontal. Butt-welded joints were used throughout and spot radiographing was required to insure the quality of the welds. On the earlier shallow tanks the inner liner was assumed to take the hydrostatic pressure of the water used for testing. A thickness of 3/8" was used for metal on recent tanks to provide increased resistance to corrosion at the liquid-vapor interface.

The inner steel tank has been used as the inner form for the pouring of the wall of the enclosing concrete tank. The steel shell alone would not withstand the external pressure of the wet concrete. However, by filling the steel tank with water and using a few horizontal circumferential stiffening rings, the thin steel shell is adequately supported and acts as the inner form as can be observed in Figure 7.

Concrete Tank

In general, the concrete tanks followed the recommendations of the Portland Cement Association Bulletin No. ST-57, "Circular Tanks without Pre-Stressing."

They were designed for full internal pressure as though the steel liner did not exist. Conventional reinforced concrete of 3,000 pounds per square inch ultimate strength (at 28 days) was used. The walls were assumed to take a load as a vertical beam partially restrained at the top and bottom and also as a cylinder loaded by ring tension. In other words, the internal pressure was resisted by both ring tension in the tank walls and by vertical beam action. This permitted thinner walls and less reinforcing steel. P.C.A. Bulletin No. ST-57 has considerable data in tabular form which aided in this analysis.

The concrete walls were also designed to resist the external earth pressure. However, this external earth pressure was not used to offset the internal hydrostatic pressure. Although the external earth pressure might prevent ultimate failure from internal pressure, it was thought that a weaker tank might expand enough to crack before the external pressure from earth would become effective.

Savings in tank construction were realized by taking advantage of two characteristic phenomena. First, the specific gravity of the waste is at its lowest when the tank is first filled and gradually increases with time during the self-evaporation process. Second, the high initial stresses in concrete caused by shrinkage during curing are gradually dissipated in time by plastic flow of the concrete. Free shrinkage of the concrete during curing is prevented by its bond to the reinforcing steel. This creates tension in the concrete.

Standard practice for the design of reinforced concrete tanks as set forth by the Portland Cement Association is based on the philosophy that the total concrete stress will be equal to the sum of the stresses developed by the shrinkage of the concrete during curing plus stresses due to the hydrostatic forces of the contained liquid whose specific gravity does not change.

A preliminary design on this basis indicated a required wall thickness of approximately 36 inches to contain liquid waste at its maximum specific gravity. But the maximum specific gravity does not occur until after concentration of the contained liquid has taken place subsequent to the initial filling of the tank.

Concrete which is subjected to stress from an external force undergoes a phenomenon known as plastic flow which continues for a period of perhaps twelve months. This plastic flow of concrete will result in a reduction of the initial shrinkage stress in the concrete. It can be said, therefore, that shrinkage stresses are at a maximum during the early life of the tank. They are at first rapidly and later gradually reduced by plastic flow in the concrete until, in a period of approximately twelve months, the shrinkage stresses in effect are relieved.

These facts resulted in two conditions of design which may be simply stated as follows:

1. The first condition appears when the tank is filled initially. The specific gravity of the liquid is at a minimum and the shrinkage stress

in the concrete is at a maximum. In this condition, the stresses due to hydrostatic force and those due to concrete shrinkage are additive.

2. The second condition occurs after a long period of storage. Self concentration of the wastes has taken place, the specific gravity of the liquid is at a maximum and the shrinkage stress has been relieved by plastic flow. In this condition, the shrinkage stress in effect may be neglected, and the design stresses may be considered as those of the applied hydrostatic liquid load.

A design developed on the basis of the above analyses indicated a required maximum wall thickness of 24 inches, in which the above second condition governed. This detail alone is estimated to have resulted in a cost saving of approximately ten per cent of the tank structure cost.

In a similar manner, savings were effected by taking into account the reduction in temperature differentials in the structure after a period of service during which the specific gravity of the contained liquid increased. Initially the temperature differentials through the structure are great and must be provided for; however, the liquid is at its lowest density at this time. Later, when the liquid density has increased, calculations indicated the temperature throughout the structure to be much more uniform. Hence a portion of the reinforcing steel originally provided for stresses resulting from temperature variations were assumed to be useful in resisting hydrostatic pressure at a later date when this pressure had increased.

Dome Roof

The dome roof is a conventional design of ellipsoidal shape built of reinforced concrete with 3,000 pounds per square inch ultimate strength. It was designed to resist an external load of six to eight feet of earth coverage plus its own weight plus a live load of two tractors at 35,000 pounds each. It has a rise to span ratio of 0.16; i.e., a rise of 12 feet in 75 feet span.

In general, the dome design followed the recommendations of the Portland Cement Association's Bulletin ST-55, "Design of Circular Domes." The ellipsoidal shape was used in order to obtain a better distribution of reinforcing steel, and to eliminate the need of a heavy edge member to carry ring tension as would be required in the case of a segmental spherical shape. The 12 foot dome rise is perhaps small, but was balanced economically against the additional excavation and backfill which would be required if a greater rise was employed.

Domed roofs, as usually employed in the design of structures, are exposed and therefore subjected to relative light loads -- dead load, wind and/or snow load. Rarely is a dome required to carry the heavy soil loading as in this particular case. The value of allowable compressive stress for concrete in domes is not specified in the American Concrete Institute Code, but past experiences have shown that compressive stresses are not high in the usual dome.

For the buried dome, however, an average load of approximately 1600 p.s.f. is not conducive to low stress. In this particular case, the compressive

stress of about 275 p.s.i. is higher than the 150-200 p.s.i. values usually encountered in domes, but considerably less than A.C.I. allowable compressive stresses in other types of members. This condition is believed reasonable and results in a dome thickness of 15 inches. The cross section of the dome near the outer edge is made thicker in order to accommodate the required amount of reinforcing steel.

The dome roof has been analyzed to determine its resistance to internal vapor pressure to assure that the anticipated operating pressures could be safely contained. Although domes of these dimensions are usually designed on the basis of carrying the applied loads by membrane stresses, it is recognized that secondary stresses will exist. It is the opinion of some authorities that ultimate failure from internal pressure will occur only from membrane stresses, i.e., that the concrete will crack and relieve the secondary bending moments before ultimate failure will take place. Cracks in the dome which will close upon cessation of pressure are not a matter of concern if they do not lead to collapse.

The secondary stresses in the dome were analyzed and found to be of insufficient magnitude to require special treatment.

APPRAISAL OF PRESENT TECHNIQUES

At present and for many years in the future, the storage of radioactive waste liquids underground in metal containers is an economical and safe method. It is realized, however, that storage in tanks has a number of limitations, including finite capacity, uncertain durability, vulnerability to accidents, and significant costs. In view of the long half life of the most hazardous isotopes such as radiostrontium, and the finite corrosion rates of the utilized metals in the stored solutions, the secure containment of liquid wastes at relatively high temperature for the necessary hundreds of years cannot be claimed as a method with absolute assurance against failure. The ultimate usable life of tanks is unknown. Although tank storage is considered an interim method of waste disposal, the technique in the present state of knowledge has the advantage that the wastes are securely under control; their location is known; and experience to date has been satisfactory. More permanent methods must eventually be provided both for existing wastes and for those of the future. Until such methods are developed the existing method can be utilized safely and can be extended economically.

FUTURE EXPECTATIONS

In order to minimize the duration of boiling and to recover certain long lived fission products such as Sr^{90} for anticipated large scale uses, a large fraction of this isotope may be removed prior to storage. The safe storage period for waste liquids may thereby be extended since the waste would exist at a lower temperature and presumably under less severe corrosion conditions. Such a development depends on the realization of a continuing, large-scale demand for the long lived fission products (Sr-90 and Cs-137). Such demand may not be realized. Furthermore, the problem of storage of the recovered isotope is still at hand. The problem is not solved; the responsibility is merely shifted elsewhere. In this way the solution of the ultimate storage problem has not been greatly advanced. The addition of any steps to the waste

confinement process can be expected to be far more expensive than the existing cheap method of tank storage. Any such cost increase must be justified on the basis of real improvement in the problem of ultimate storage.

Future objectives directed at the solution of the problem are expected to be along the lines of processes which will convert liquid wastes into inert solids by some calcination technique. Such processing may be conducted after a storage period of more than two years during which time the heat generation potential which has been reduced by a factor of at least ten will present less of an engineering problem in the design of facilities for its dissipation. The inert solids which would then have low heat producing potential might then be packaged for permanent storage in dry vaults, pits, caves, abandoned mines, etc. Such methods have been under study at several sites for some time. The problems are essentially those of engineering and development of the most economical and reliable special processing facilities that would be required.

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(1) Coleman, Allan T., et al, Threshold Limit Values for 1957, American Medical Association Archives of Industrial Health, 16:261-265, (September, 1957).

(2) United States Department of Commerce, National Bureau of Standards, Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water, Handbook 52, pages 8 and 15.

(3) Cook, M. W. and Waters, E. D., General Electric Co., Hanford Atomic Products Operation, Richland, Washington, Operational Characteristics of Submerged Gas-Lift Circulators, HW-39432, (12/1/55)

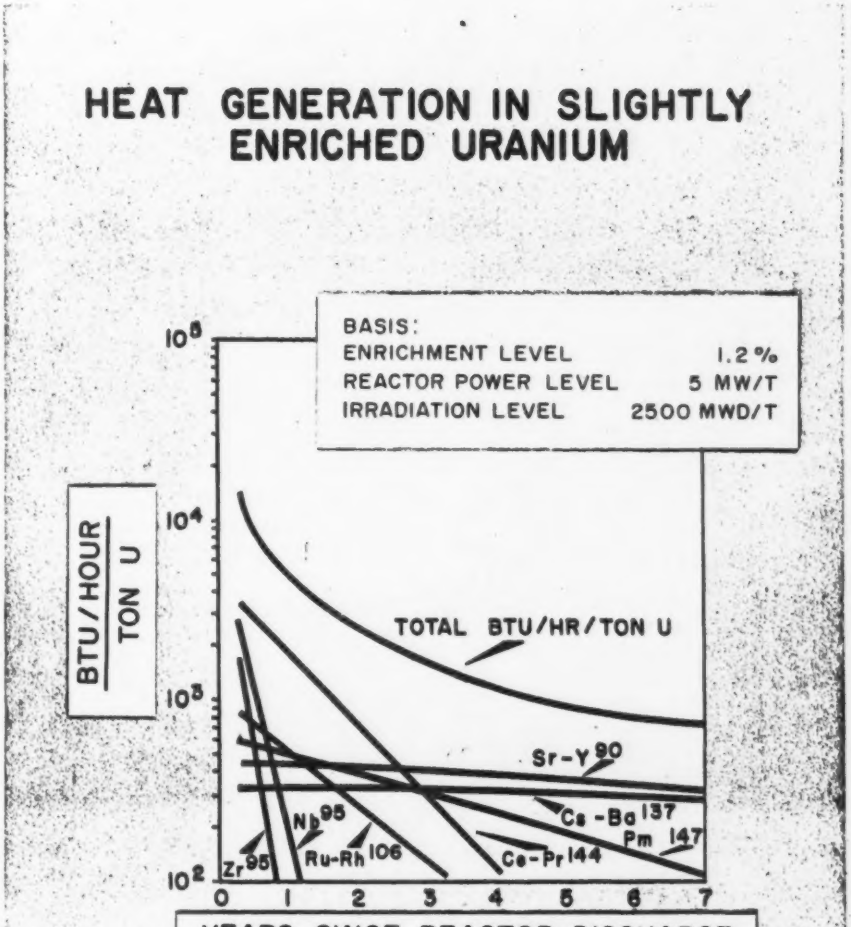

HEAT GENERATION IN SLIGHTLY ENRICHED URANIUM
BASIS:
ENRICHMENT LEVEL 1.2 %
REACTOR POWER LEVEL 5 MW/T
IRRADIATION LEVEL 2500 MWD/T
BTU/HOUR / TON U
10^5
10^4
10^3
10^2
TOTAL BTU/HR/TON U
Sr-Y 90
Cs-Ba 137
Pm 147
Nb 95
Ru-Rh 106
Ce-Pr 144
Zr 95
0 1 2 3 4 5 6 7
YEARS SINCE REACTOR DISCHARGE

Figure 1

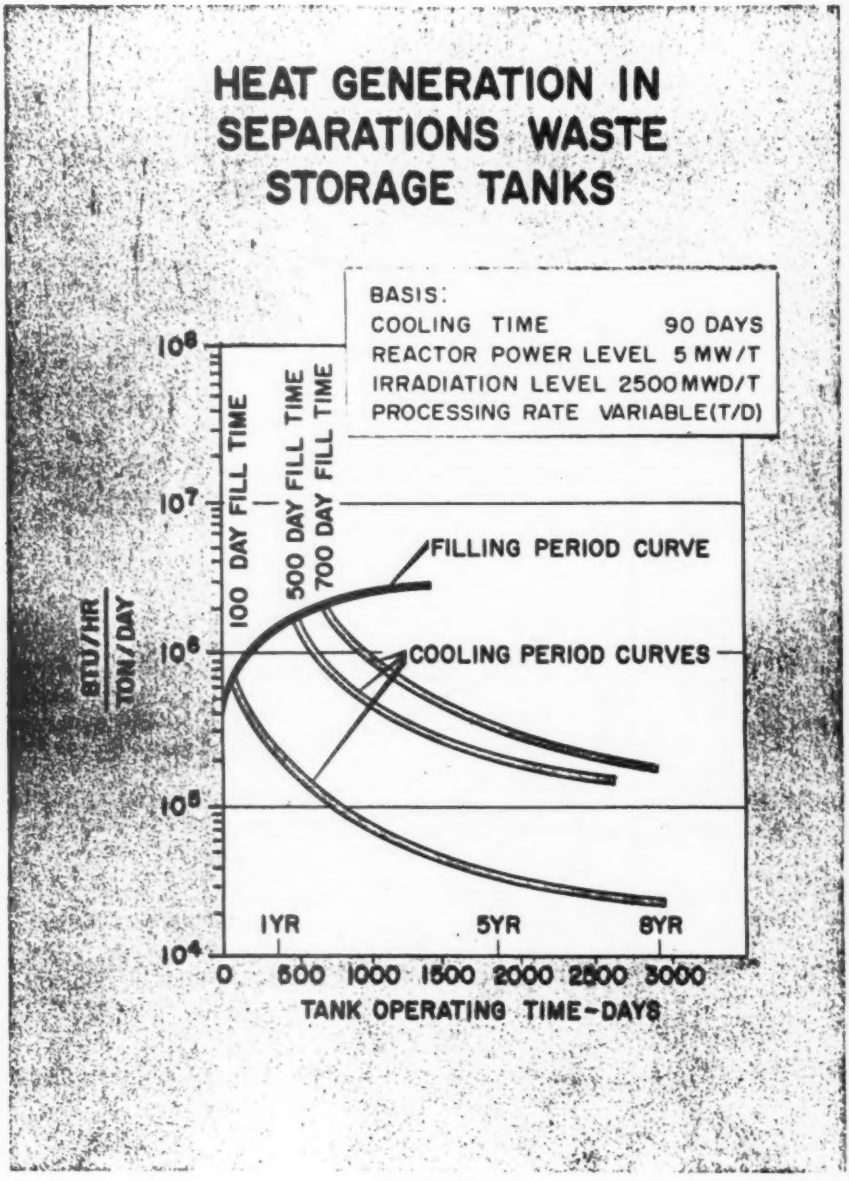
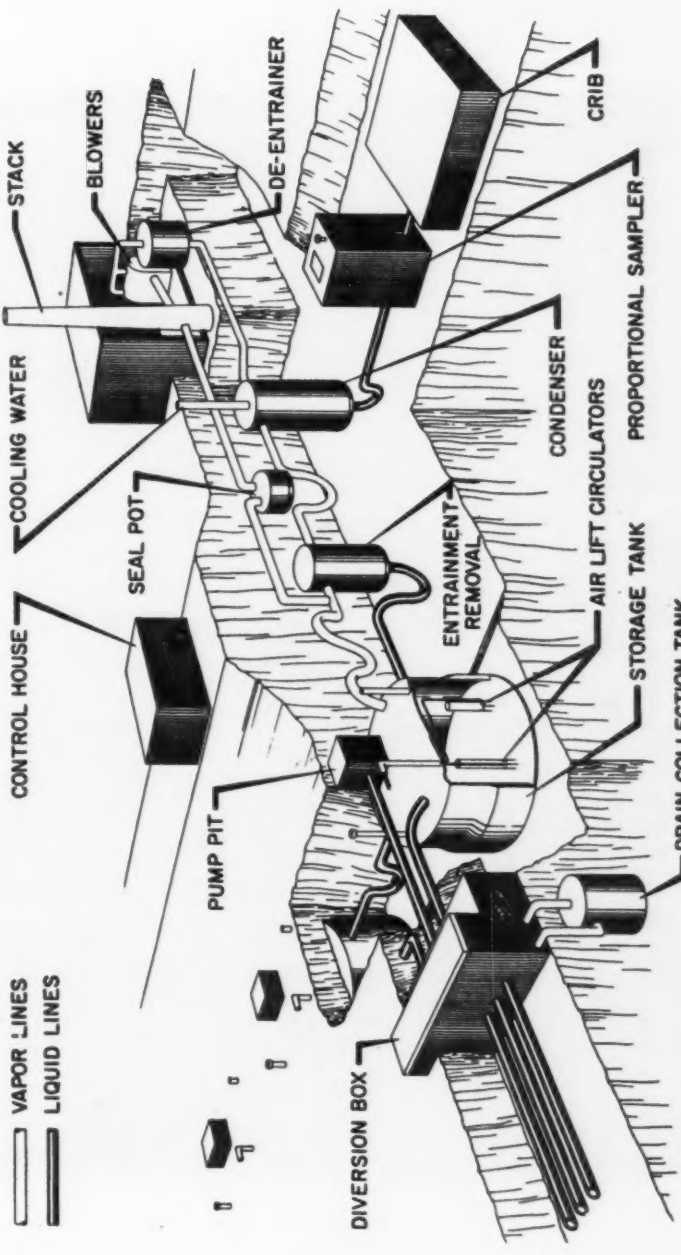

HEAT GENERATION IN SEPARATIONS WASTE STORAGE TANKS
BASIS:
COOLING TIME 90 DAYS
REACTOR POWER LEVEL 5 MW/T
IRRADIATION LEVEL 2500 MWD/T
PROCESSING RATE VARIABLE(T/D)
100 DAY FILL TIME
500 DAY FILL TIME
700 DAY FILL TIME
FILLING PERIOD CURVE
COOLING PERIOD CURVES
BTU/HR / TON/DAY
10⁸
10⁷
10⁶
10⁵
10⁴
1YR
5YR
8YR
0 500 1000 1500 2000 2500 3000
TANK OPERATING TIME-DAYS

Figure 2

FIGURE 3
WASTE STORAGE FACILITY ARRANGEMENT



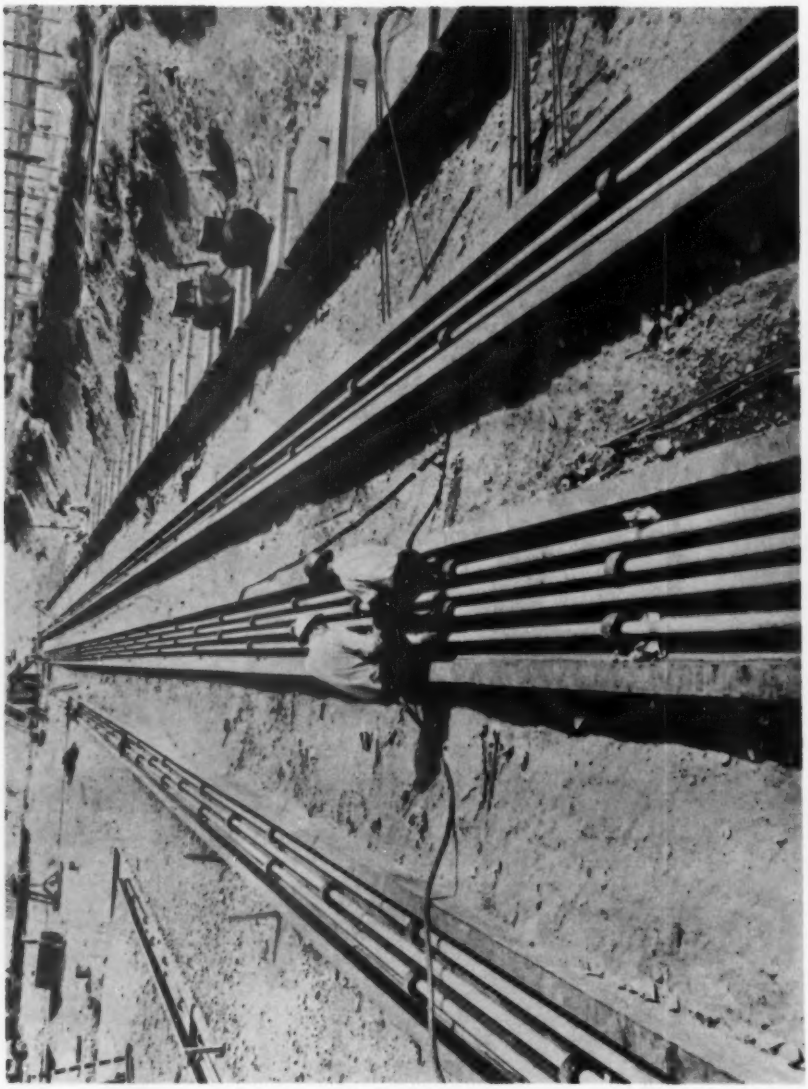


Figure 4

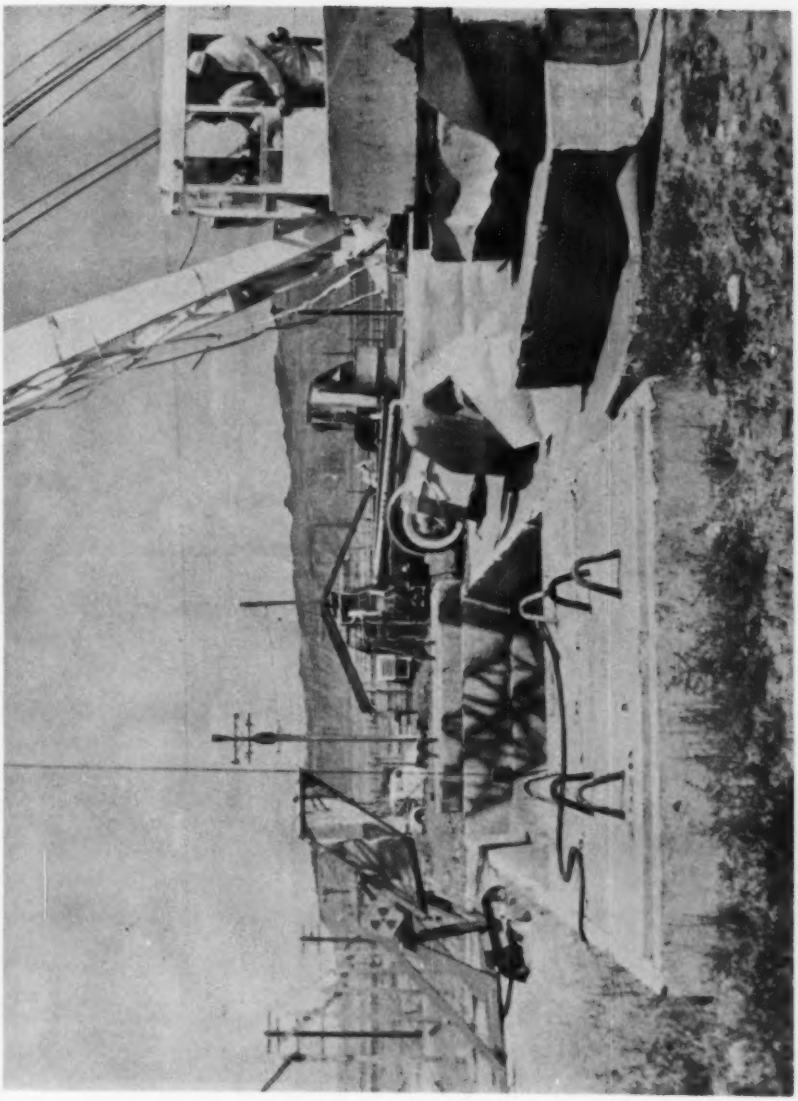


Figure 5

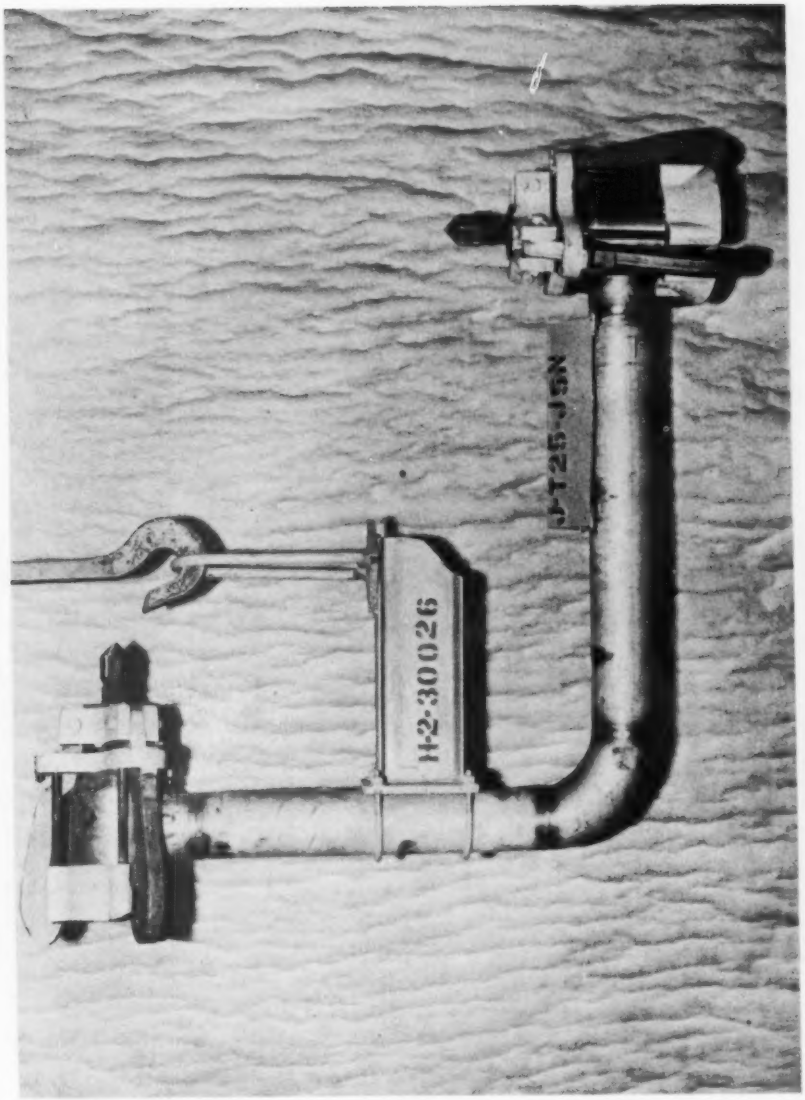


Figure 6

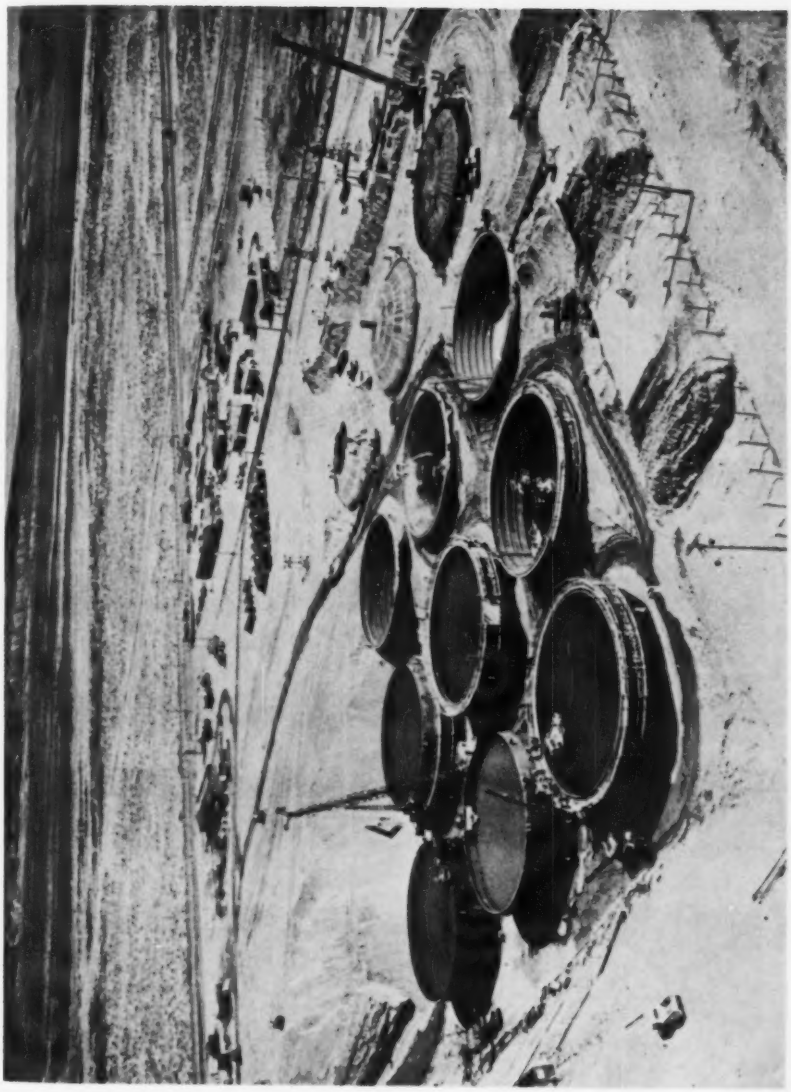


Figure 7

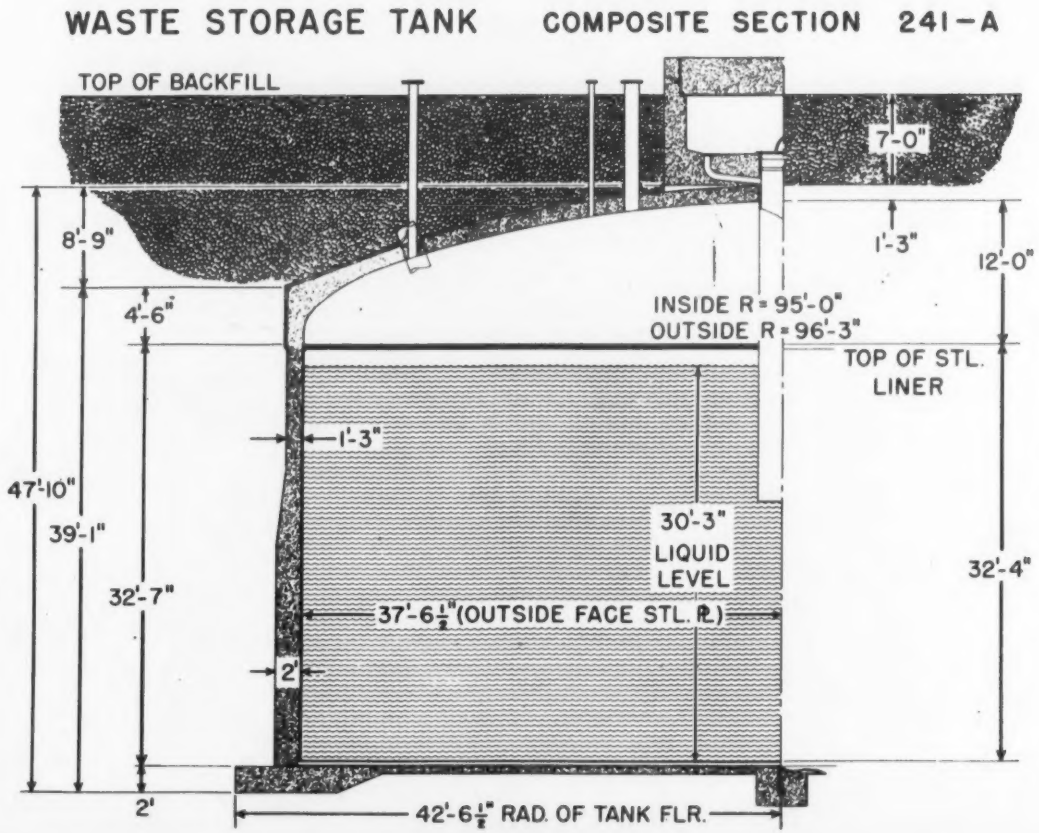

WASTE STORAGE TANK COMPOSITE SECTION 241-A
TOP OF BACKFILL
7'-0"
8'-9"
1'-3"
12'-0"
4'-6"
INSIDE R = 95'-0"
OUTSIDE R = 96'-3"
TOP OF STL. LINER
1'-3"
47'-10"
39'-1"
30'-3" LIQUID LEVEL
32'-4"
32'-7"
37'-6½" (OUTSIDE FACE STL. PL.)
2'
2'
42'-6½" RAD. OF TANK FLR.

Figure 8

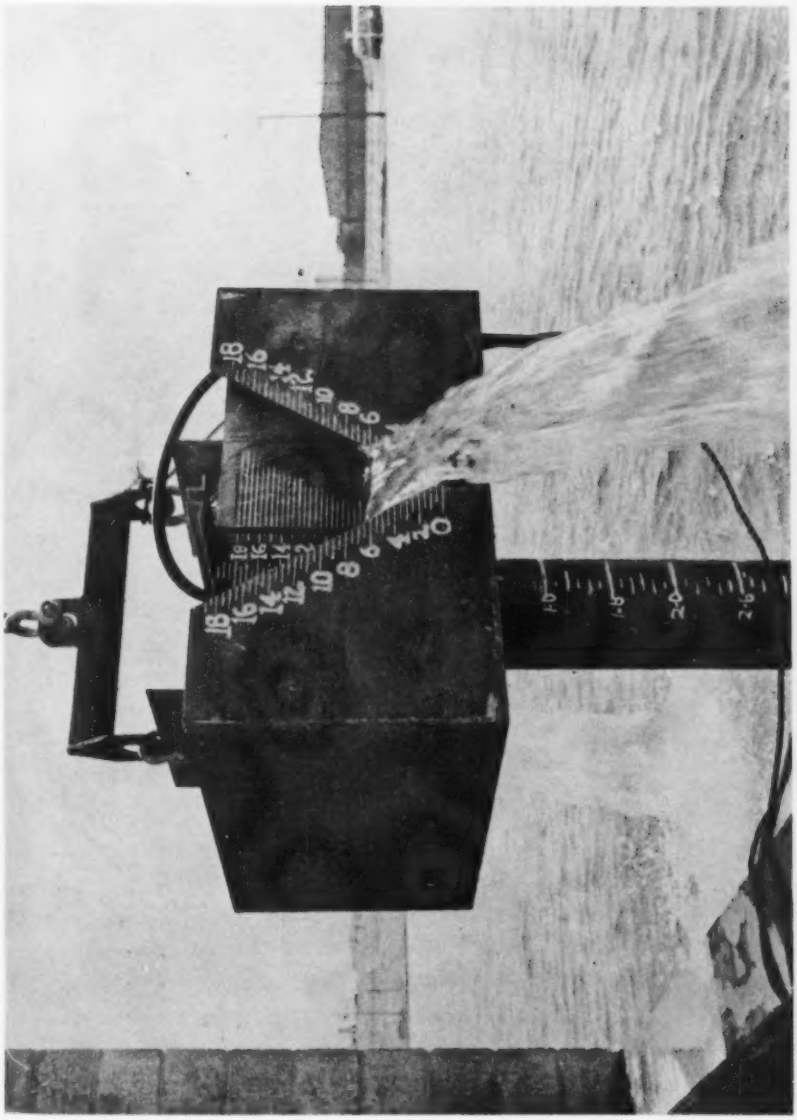


Figure 9

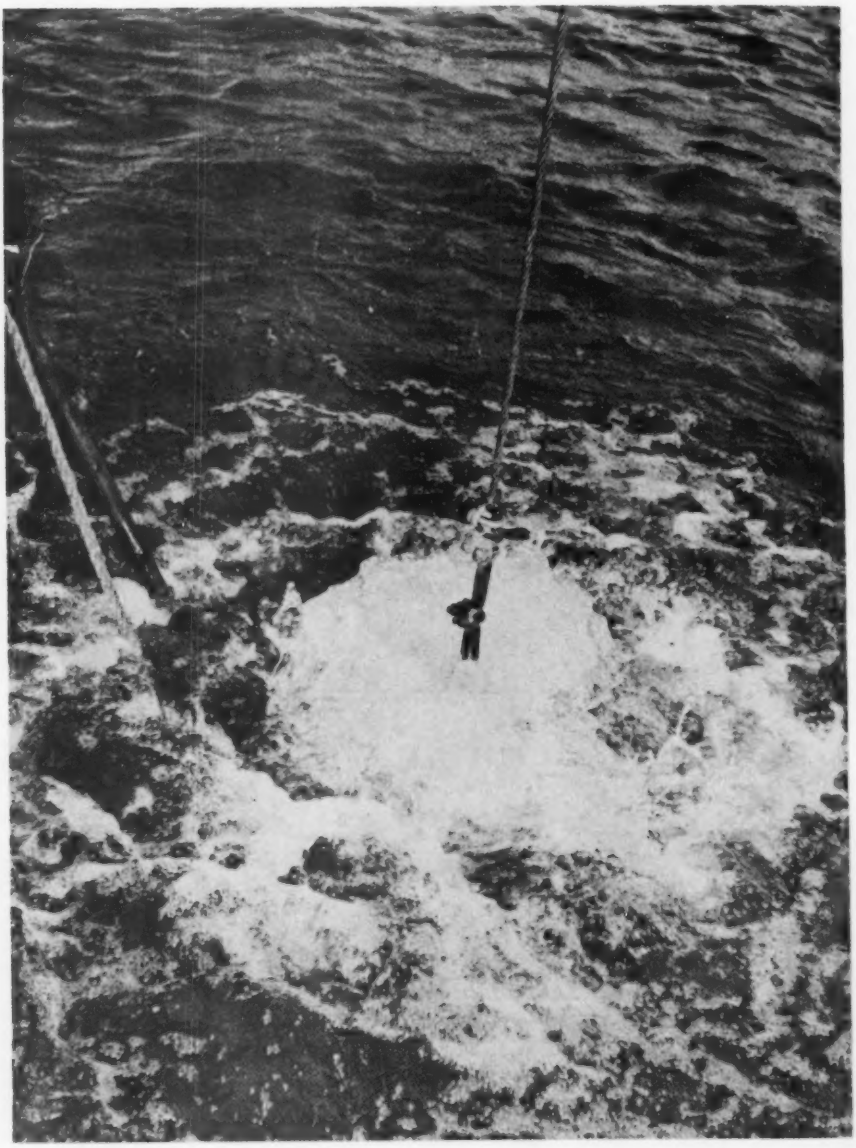


Figure 10

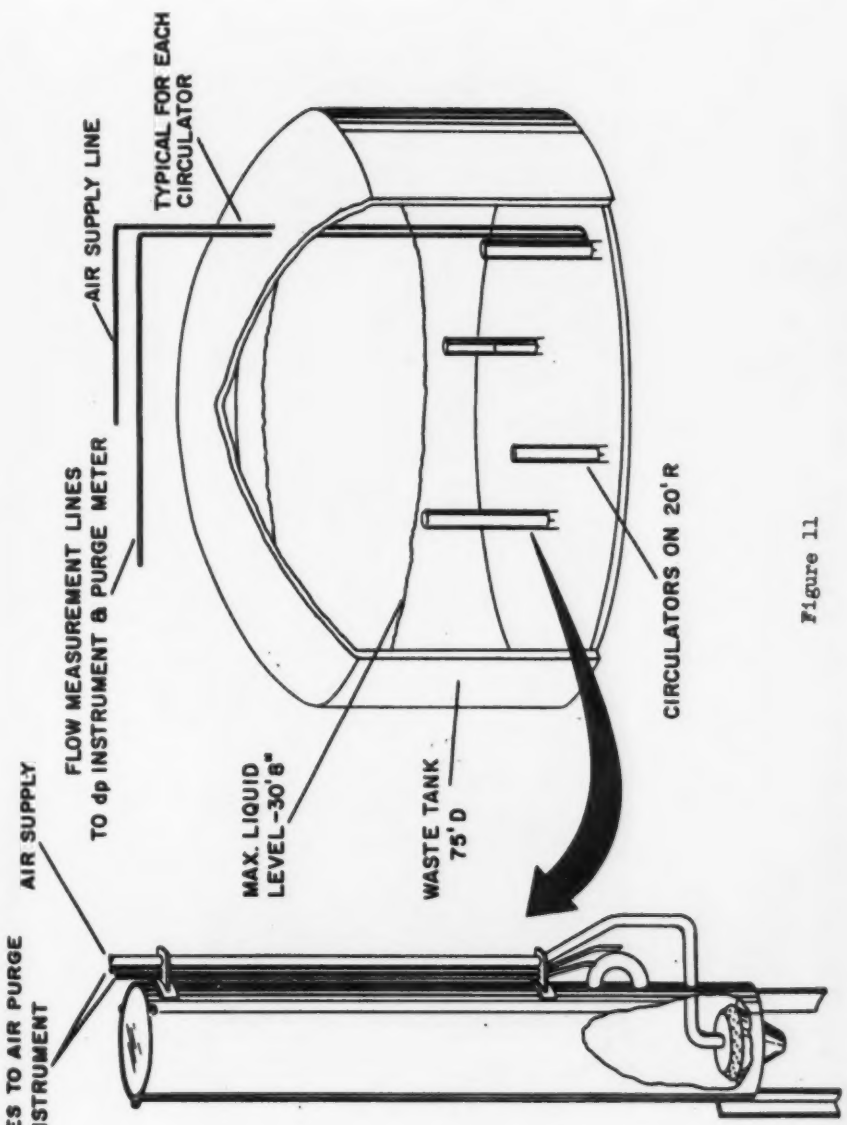

dp LINES TO AIR PURGE & dp INSTRUMENT
AIR SUPPLY
FLOW MEASUREMENT LINES TO dp INSTRUMENT & PURGE METER
AIR SUPPLY LINE
TYPICAL FOR EACH CIRCULATOR
MAX. LIQUID LEVEL -30'8"
WASTE TANK 75'D
CIRCULATORS ON 20'R

Figure 11

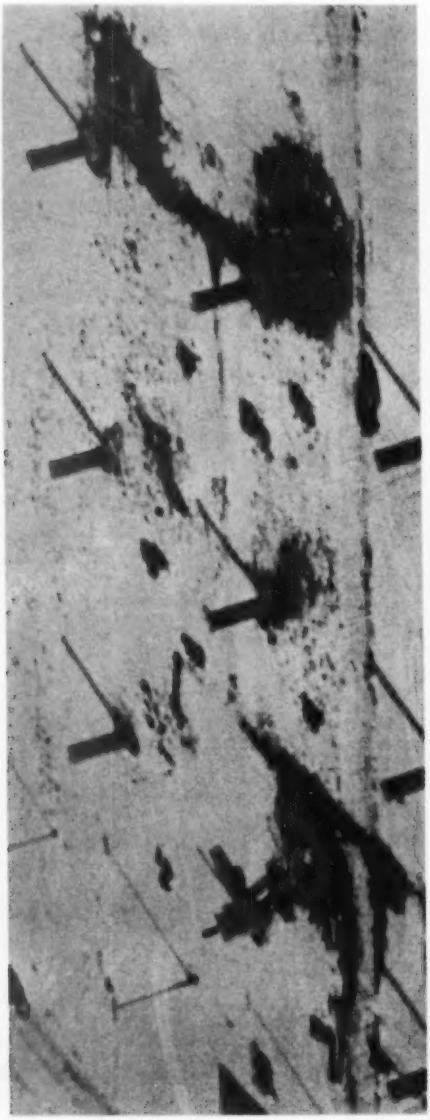


Figure 12

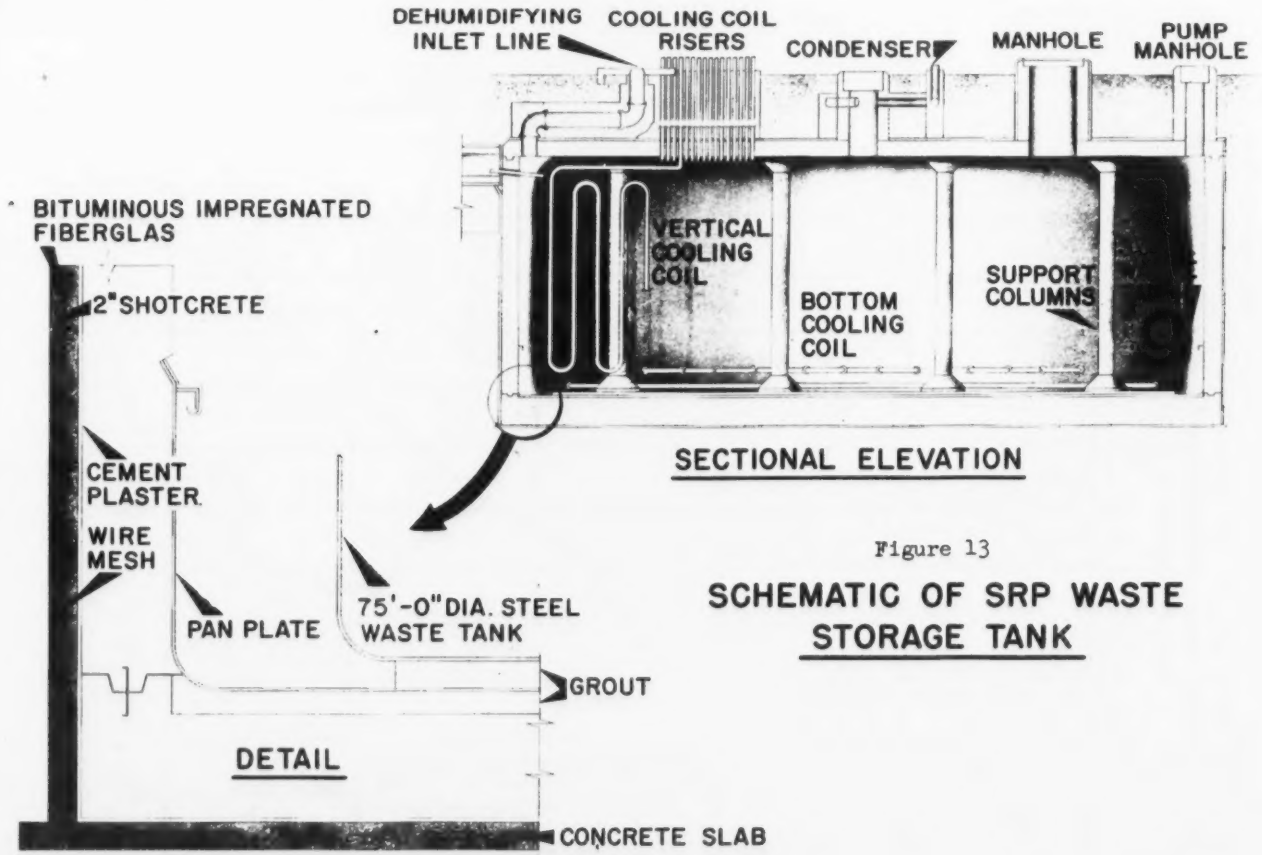


Figure 13

SCHEMATIC OF SRP WASTE STORAGE TANK

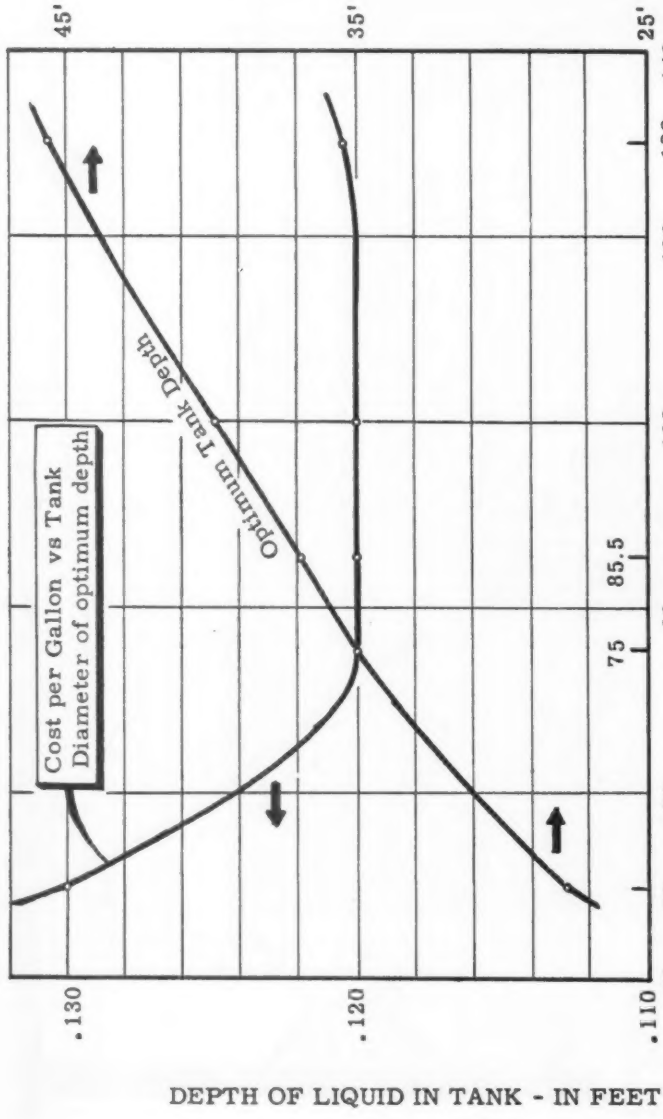

COST PER GALLON OF STORAGE - IN DOLLARS
Included: Bare tanks, excavation & backfill - Contract price
Not Included: Piping, condensers, accessories, engineering and overhead.
45'
35'
25'
Cost per Gallon vs Tank Diameter of optimum depth
Optimum Tank Depth
.130
.120
.110
40
50
60
75
80
85.5
100
120
130
140
Diameter of Tank in feet
Optimum Dimensions
UNDERGROUND WASTE STORAGE CONTAINERS
VERTICAL CYLINDRICAL TANKS WITH ELLIPSOIDAL DOME ROOFS
DEPTH OF LIQUID IN TANK - IN FEET

Figure 14

Representative HOLIFIELD. We are running a little late, and if it is all right with the committee and with Dr. Morgan, we will ask him to be our first witness tomorrow morning rather than at this time.

We will have hearings both in the morning and in the afternoon tomorrow. We have a long day ahead of us tomorrow.

Thank you very much.

(Whereupon, at 12:30 p.m., Wednesday, January 28, 1959, the hearing was recessed, to reconvene the following day, Thursday, January 29, 1959, at 10 a.m. in room P–63 of the Capitol.)

Figure 14

INDUSTRIAL RADIOACTIVE WASTE DISPOSAL

THURSDAY, JANUARY 29, 1959

CONGRESS OF THE UNITED STATES,
SPECIAL SUBCOMMITTEE ON RADIATION,
JOINT COMMITTEE ON ATOMIC ENERGY,
Washington, D.C.

The special subcommittee of the Joint Committee on Atomic Energy met at 10 o'clock a.m., pursuant to recess, in the Old Supreme Court Chamber, the Capitol, Hon. Chet Holifield, chairman of the special subcommittee, presiding.

Present: Representatives Holifield, Durham (chairman of the committee), Price, Hosmer, Bates; Senators Pastore, Hickenlooper, and Dworshak.

Also present: James T. Ramey, executive director, David R. Toll, staff counsel, and Dr. Walton Rodger, technical adviser, Joint Committee on Atomic Energy.

Representative HOLIFIELD. The committee will be in order.

This morning the Special Subcommittee on Radiation will continue its hearings on industrial radioactive waste disposal.

Today we are scheduled to receive statements concerning waste management operations at Oak Ridge, at the National Reactor Testing Station at Idaho, and at the Shippingport pressurized water reactor.

After that we will begin to explore the waste disposal research and development program, with statements from scientists and technical people who are doing work in this field.

Today we have a full day. We hope to hear from about six or seven witnesses this morning, and five or six more this afternoon.

I will have to request each witness, therefore, to stay within the limits of time allotted on the outline.

This, of course, is always modified by the right of the committee members to cause them to go over that time by asking questions.

I would like to also request each witness to use lay language, understandable to those of us who have not had the benefit of your technical training and many years of experience in this field.

All technical material will be included in the public record, but the subcommittee will appreciate it if the oral summaries are presented briefly and in language understandable to the layman and the public.

I would also like to announce that the hearings tomorrow morning will be held in room F–39, of the Capitol, which is on the floor beneath.

Our first witness this morning is Dr. K. Z. Morgan of the Health Physics Division of Oak Ridge National Laboratory.

We are very glad to have you with us this morning, Dr. Morgan. Would you please come forward and proceed.

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I am sorry we could not get to you yesterday, but we are glad to lead off with you this morning.

Chairman DURHAM. Doctor, we are very glad to have you with us. He is a former fellow townsman of mine. We are glad to have you back with us.

STATEMENT OF K. Z. MORGAN,[1] DIRECTOR OF THE HEALTH PHYSICS DIVISION, OAK RIDGE NATIONAL LABORATORY

Dr. MORGAN. Thank you, sir.

The two primary objectives in waste management are radiation protection and economy. Radiation protection requirements are based upon recommendations of the National Committee on Radiation Protection—NCRP—and the International Commission on Radiological Protection—ICRP. The economics of waste disposal are of primary concern if atomic energy is to compete with present industrial power. Perhaps it should be emphasized that safety and economy are not independent entities for there is no denying that health physics protection is expensive; however, the final cost of an operation can be increased exceedingly if we fail to provide sufficient health physics protection.

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[1] Born in Kannapolis, N.C., in 1907. He received the A.B. degree in 1929 and the M.A. degree in 1930 from the University of North Carolina. He worked with the Westinghouse Electric Corp. in 1931 and received the Ph. D. in physics from Duke University in 1934. From 1934 to 1943 he was chairman of the physics department of Lenior Rhyne College, and during this period he did research work in cooperation with Duke University in the field of cosmic radiation. This led to a number of cosmic ray expeditions, among which were measurements in Linville Caverns, on the top of Mount Mitchell, and on Beech Mountain (all in North Carolina), and on Mount Evans near Denver, Colo.

He became a member of the Metallurgical Laboratory staff at the University of Chicago in the spring of 1943. Here he was one of a small group to develop and establish the new science known as health physics, which today employs over 1,000 persons in this country alone. Health physics has the responsibility of establishing acceptable levels of maximum permissible exposure, developing instruments to make measurements to ascertain that safe exposure levels are maintained, and making a study of attenuation of radiation matter in order to determine the effectiveness of shielding material for radiation protection and in order to estimate more accurately the dose delivered by this radiation at various depths in the human body.

During the fall of 1943 Dr. Morgan transferred to the Oak Ridge National Laboratory, where to the present he is Director of the Health Physics Division, which employs over 150 persons who are engaged in research, engineering, and applied problems. In 1957 he was the first president of the Health Physics Society, and at present he is a member of the board of the society and editor of the journal Health Physics. He is a member of a number of committees dealing with problems of radiation protection, such as the Committee of the National Research Council on Shipment of Radioisotopes and the main committees of the International Commission on Radiological Protection and the National Committee on Radiation Protection. He is Chairman of the National and International Committees on Maximum Permissible Internal Dose of Radioisotopes. He has published a number of papers dealing with subjects of cosmic rays, radiation protection, instrumentation, and general health physics.

One of the current problems of interest is the study and development of suitable methods of radioactive disposal that can be applied when atomic energy reactors are widely used as a source of electrical power. The Health Physics Division, of which he has charge, has recently made a number of important engineering advances in the study of disposal of radioactive waste with respect to both liquid and gaseous contaminants. This same division has led the way in the training of health physicists who, in turn, have established health physics programs in many universities, hospitals, and industrial laboratories. The Oak Ridge Health Physics Division has developed the first instruments with which quantitative measurements of fast neutron dose can be made, and it is now in the process of measuring many physics constants useful in estimating radiation damage, such as the electron volts per ion pair, stopping power, and scattering of ionizing radiation as it passes through matter. In 1953 the National Committee on Internal Dose, of which Dr. Morgan is chairman, released the first internal dose report, listing the maximum permissible body burden and maximum permissible concentration of some 70 radioisotopes. The following year another committee of which he is chairman released for publication by the International Commission on Radiological Protection a "Maximum Permissible Internal Dose Report" extending these calculations to other radioactive isotopes. Both the above internal dose reports have just been revised and submitted (January 1959) for publication.

He is a member of the American Association for the Advancement of Science, the American Industrial Hygiene Association, the American Nuclear Society, Research Society of America, the Radiation Society, the Tennessee Academy of Science, associate fellow of the American College of Radiology, a fellow of the American Physical Society, and a member of the board of the Health Physics Society.

There are two basic methods generally in use for the disposal of radioactive waste:

1. The waste may be concentrated and the concentrate stored; or
2. The waste may be combined with a passive medium and dispersed throughout the atmosphere, soil and/or water, such that its concentration is reduced to acceptable levels.

Fortunately, the Oak Ridge National Laboratory is located in an area which affords several local advantages, as follows:

1. The relative isolation of the laboratory;
2. The large volume of water available in the Clinch River, which partly surrounds the Oak Ridge area; and
3. The extensive Conasauga shale formation.

The boundaries of the Atomic Energy Commission controlled area are, at the closest point, about 2 miles from the Laboratory. The nearest large city, Knoxville, Tenn., is approximately 20 miles from the controlled area. Surveys have shown that all the surface and ground water drainage from the Laboratory, and waste disposal area, leads to an impoundment fed by White OakCreek, which flows to the Clinch River. The White Oak-Clinch River drainage system has been monitored for radioactive contamination since the beginning of operations at the Laboratory in 1943. Dilution afforded by the Clinch River lowers liquid waste concentration to acceptable levels. The Clinch River flows into the Tennessee River 20.8 miles below the outfall of White Oak Creek, and both rivers form a part of the TVA system of rivers and reservoirs. Thus, in case of an emergency, TVA may be called upon to adjust the flow of the river system to provide holdup and/or dilution.

Representative HOLIFIELD. Has there been any request that the TVA adjust their flow?

Dr. MORGAN. We have had discussions with TVA and they have agreed they can work with us in case of an emergency.

Representative HOLIFIELD. They have not been called upon as yet?

Dr. MORGAN. That is correct.

Representative HOLIFIELD. When you use White Oak Lake as part of the system it did reach equilibrium, I understand, and you abandoned storage at White Oak Lake?

Dr. MORGAN. That is correct.

Representative HOLIFIELD. Is there any possibility that the seepage of the basin system you are now using might likewise reach equilibrium and might be abandoned?

Dr. MORGAN. Yes; but at the rate of travel of the long-lived radionuclides we would expect this to be a hundred years or more.

Seepage pits provide a safe and relatively cheap means of disposal for a significant fraction of the liquid and solid radioactive waste from the Laboratory. The Conasauga shale formation allows for seepage pits as the shale has the ability to capture and hold certain of the radioactive components while allowing water to seep through.

The simplified diagram we have (figure 1) shows the type of wastes that are handled and the methods used for their disposal. It is apparent that the Oak Ridge National Laboratory is disposing of waste by discharging into the atmosphere, by holdup in tanks, by disposal into the soil, and by seepage into the Clinch River system. The continuing goal of the laboratory is to immobilize as much of the radio-

active materials as possible by fixation in the soil or as insoluble sludge in the tanks and pits and to hold to a minimum the amount of radioactive material discharged to the river and to the atmosphere.

You will note the flow of highly radioactive chemical wastes [pointing to 1 in fig. 1] which consists of 7,000 gallons per day and this particular waste at the Laboratory has a concentration of radioisotopes of one one-thousandth to one one-hundredth curies per gallon, while at large production plants the concentration may be several thousand curies per gallon. I mention this not to minimize our waste disposal problems—since we are aware that even small amounts of some of the radioisotopes when improperly handled can be extremely hazardous—but to give you some yardstick for comparing our disposal problems to those of the production plants.

Most of the highly radioactive waste is produced in chemical processing buildings and laboratories. It is discharged from laboratory sinks and process drains to stainless steel pipes and to underground stainless steel monitoring tanks as shown here [pointing to the 19 monitoring tanks shown under 1 in fig. 1]. The rate of waste accumulation is continuously recorded at one location through a telemetering system and the tank volumes are manually checked every 4 hours.

When a monitoring tank fills the waste is pumped to three underground concrete storage tanks where it is temporarily stored to allow the short-lived radioisotopes to decay before it is pumped to the waste pits. These waste pits are located 1½ miles from the Laboratory. The waste pits are three 1-million-gallon capacity, open pits excavated in the earth in a location chosen for its remoteness from the Laboratory, the type of soil and the natural underground water drainage to White Oak Creek. This particular soil, made up of clay and Conasauga shale, removes and retains the radioisotopes while the solution seeps slowly through it toward White Oak Creek.

The liquid uranium waste (shown under 2 in fig. 1) is also a product of the chemical processing operations and research. This type of waste is presently produced in very small quantities averaging no more than a hundred gallons per week and is not a true waste in the strict sense of the word, since it is a solution of reusable uranium contaminated with fission products. This material is collected and stored in underground stainless steel monitoring tanks where a check is kept on the sources of waste and their volumes. When a tank is filled the waste is pumped to underground concrete storage tanks [pointing to the 42,000-gallon and 170,000-gallon tanks under 2 in fig. 1], where it is held until transfer to a small processing plant for uranium recovery.

The 700,000 gallons per day of mildly contaminated process water [pointing to 3 in fig. 1] is the least radioactive of all the Laboratory waste, yet it is the most difficult to manage because of its large volume. It is made up mainly of water used for cooling equipment and water from floor drains and under normal conditions carries a radioactive contamination of only one ten-thousandth of that handled by the highly radioactive chemical waste system. A considerable amount of the radioactive liquid put through this system is the result of equipment failure, human error, or accidents. Whenever these incidents

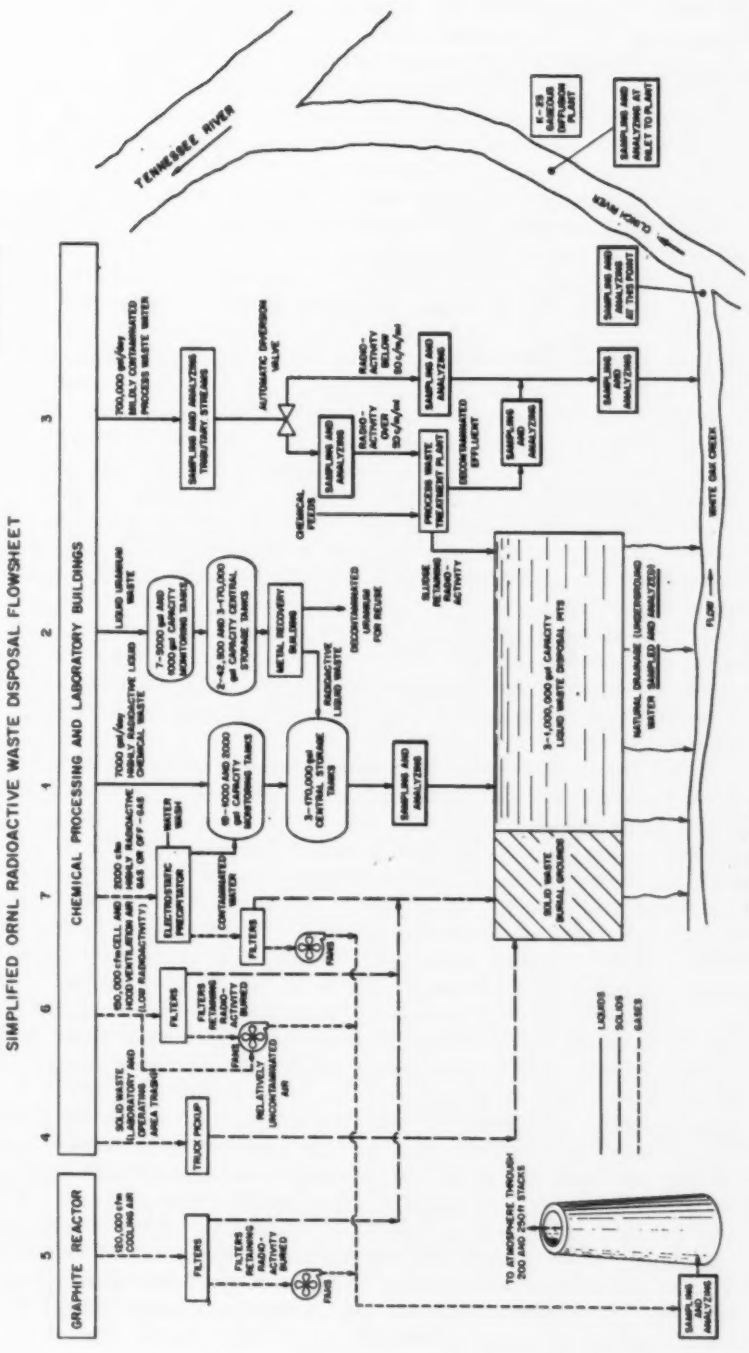

SIMPLIFIED ORNL RADIOACTIVE WASTE DISPOSAL FLOWSHEET
CHEMICAL PROCESSING AND LABORATORY BUILDINGS
GRAPHITE REACTOR
TENNESSEE RIVER
CLINCH RIVER
WHITE OAK CREEK
WHITE OAK DAM
FLOW
SOLID WASTE (LABORATORY AND OPERATING AREA TRASH)
TRUCK PICKUP
SOLID WASTE BURIAL GROUNDS
FILTERS
FANS
FILTERS RETAINING RADIO-ACTIVITY BURIED
RELATIVELY UNCONTAMINATED AIR
150,000 cfm CELL AND HOOD VENTILATION AIR (LOW RADIOACTIVITY)
2000 c/m HIGHLY RADIOACTIVE GAS OR OFF-GAS
ELECTROSTATIC PRECIPITATOR
WATER WASH
CONTAMINATED WATER
7000 gal/day HIGHLY RADIOACTIVE CHEMICAL WASTE
LIQUID URANIUM WASTE
700,000 gal/day HIGHLY CONTAMINATED PROCESS WASTE WATER
SAMPLING AND ANALYZING TRIBUTARY STREAMS
AUTOMATIC DIVERSION VALVE
CHEMICAL FEEDS
SAMPLING AND ANALYZING
RADIO-ACTIVITY OVER 50 c/s/ml
RADIO-ACTIVITY BELOW 50 c/s/ml
PROCESS WASTE TREATMENT PLANT
DECONTAMINATED EFFLUENT
SLUDGE RETAINING RADIO-ACTIVITY
6-1000 AND 2000 gal CAPACITY MONITORING TANKS
7-5000 gal AND 1000 gal CAPACITY MONITORING TANKS
2-42,300 AND 3-170,000 gal CAPACITY CENTRAL STORAGE TANKS
3-170,000 gal CENTRAL STORAGE TANKS
METAL RECOVERY BUILDING
RADIOACTIVE LIQUID WASTE
DECONTAMINATED URANIUM FOR REUSE
3-1,000,000 gal CAPACITY LIQUID WASTE DISPOSAL PITS
NATURAL DRAINAGE (UNDERGROUND) WATER SAMPLED AND ANALYZED
SAMPLING AND ANALYZING AT THIS POINT
SAMPLING AND ANALYZING AT INLET TO PLANT
K-25 GASEOUS DIFFUSION PLANT
TO ATMOSPHERE THROUGH 200 AND 250ft STACKS
SAMPLING AND ANALYZING
LIQUIDS
SOLIDS
GASES

FIGURE 1

take place an immediate effort is made to correct the conditions in order to direct the radioactive material to the high level system.

Because of the large quantity of process waste generated per day, the storage of it for any appreciable time is impractical and leaves the Laboratory no choice other than to dispose of it on a current basis. At one time when the levels of radioactive material handled by the Laboratory were relatively low, the low level waste was discharged to the creek without treatment and comparatively little effort was required in monitoring the waste streams before discharge to the creek. During the last 13 years as the Laboratory expanded and as the levels of radioactive material increased by a factor of about 100, methods had to be developed for monitoring all the waste streams and for removing most of the radioisotopes from the waste before it was discharged to the creek.

Control of the discharge to the creek is accomplished mainly by frequent samplings and monitoring of the various waste streams and their combined waste flow before its passage through the automatic diversion valve. If an analysis of the combined waste indicates a substantial increase in radioactivity, samples of the individual streams are analyzed to determine the source and inform the personnel whose operations are generating excessive radioactive waste. Sometimes personnel are not aware that their operations are generating excessive radioactive waste and thus it is necessary to contact them and make appropriate corrections in their operations.

All the waste generated during the regular working hours is collected and treated before discharge to the creek. During the off-shift hours, however, the waste containing less than 50 counts/min/ml (β,γ) is diverted to the creek through a settling basin, which discharges directly to the creek.

From 53 to 87 percent of the total radioactivity contained in the waste diverted into the process treatment plant is removed by a single pass through the plant; 73 to 87 percent of strontium 90, considered to be the most hazardous component, is removed. The process employed in this plant for the treatment of radioactive waste is new in atomic energy operations. It is the first plant built that utilizes on a large scale a standard, but modified lime-soda ash process normally used for water softening. The plant has been in operation for only about a year and many variations of the process remain to be tested; therefore its full efficiency may not yet be realized. It is probable that this plant will serve as a model for other atomic energy installations having the problem of disposal of large volumes of low level radioactive waste.

Practically all the solid waste [pointing to 4 in fig. 1] produced at the Laboratory consists of paper, glassware, and other pieces of laboratory equipment, much of which is contaminated with radioactive material. As compared to the liquid waste, the solid wastes usually contain an almost insignificant quantity of radioactive material, probably about 1 percent of the total disposed by the Laboratory. Accurate determination of the quantity of radioisotopes in the solid waste is difficult because the waste is composed of a nonuniform mixture of many materials.

In addition to being used for disposal of laboratory wastes, the burial grounds are used for disposal of wastes from other AEC installations and contractors and the Laboratory's radioisotope customers. Radioisotope customers contribute a comparatively insignificant volume; other installations, mainly Argonne National Laboratory, Knolls Atomic Power Laboratory, Mound Laboratories, Battelle Memorial Institute, and General Electric in Evansdale, Ohio, account roughly for half the land area assigned permanently to this burial program. To date, 25 acres have been used by the burial ground operation. The current rate of use is approximately 5 acres per year.

Before reviewing the waste gas disposal operations, it is necessary to give some explanation of the term "waste gas." Actually, a more descriptive term would be "contaminated air" since the bulk of the waste is composed of air that is contaminated with radioactive materials. The airborne radioactive contaminants may consist of either gaseous or microscopic solid particles, usually a combination of both.

Because of extremely large volumes it is impractical to store most of the gaseous waste for any appreciable period of time. The only practical method of disposal must, therefore, involve continuous decontamination of the gases as they are produced and discharged to the atmosphere.

The 120,000 cubic feet per minute of air used for cooling the graphite reactor [pointing to 5 in fig. 1] is drawn through the reactor before going to the 200-foot stack and passes through filters that remove most of the radioactive particles which if discharged to the atmosphere would create the principal hazard from airborne radioactive contamination.

The cell and hood ventilation system is a network of ducts which collects under low vacuum a relatively large volume of exhaust air from the laboratory hoods and concrete shielded cells, shown here [pointing to 6 in fig. 1]. The air is drawn through the system by five electrically driven fans having capacities ranging from 25,000 to 60,000 cubic feet per minute; in case of a failure of the electrical equipment, three steam driven fans are used. The air from this system is discharged into the atmosphere through two stacks, one 200 feet and the other 250 feet high.

The chemical processing of radioactive material usually results in the release of highly concentrated radioactive contaminants in a relatively small volume of gas that is handled by the off-gas system [pointing to 7 in fig. 1]. The gases are collected and drawn by vacuum through a network of underground stainless steel lines and are discharged after cleanup through the 250-foot stack.

This off-gas system is the primary radioactive gas disposal facility, while the cell ventilation system [pointing to 6 in fig. 1] is the secondary system for all large scale chemical processing. The off-gas system is connected directly to the processing vessels, whereas the cell ventilation system is connected only to the cell where it removes the air from the area in which the vessels are located. Unless there is a failure all the gaseous activity is removed by the off-gas system keeping the air drawn into the cell ventilation system free from contamination. This arrangement reduces the volume of gas that must be decontaminated.

The decontamination of the gases handled by the off-gas system is accomplished partly in the operating cells by scrubbing the air with caustic and passing it through an electrostatic precipitator and filters before it goes to the off-gas system.

Let us now examine the effectiveness of the waste management system of our Laboratory.

Representative HOLIFIELD. Do you monitor your gases?

Dr. MORGAN. The gas is monitored in each of the stacks before leaving the Laboratory and then it is monitored in the Laboratory and in the area.

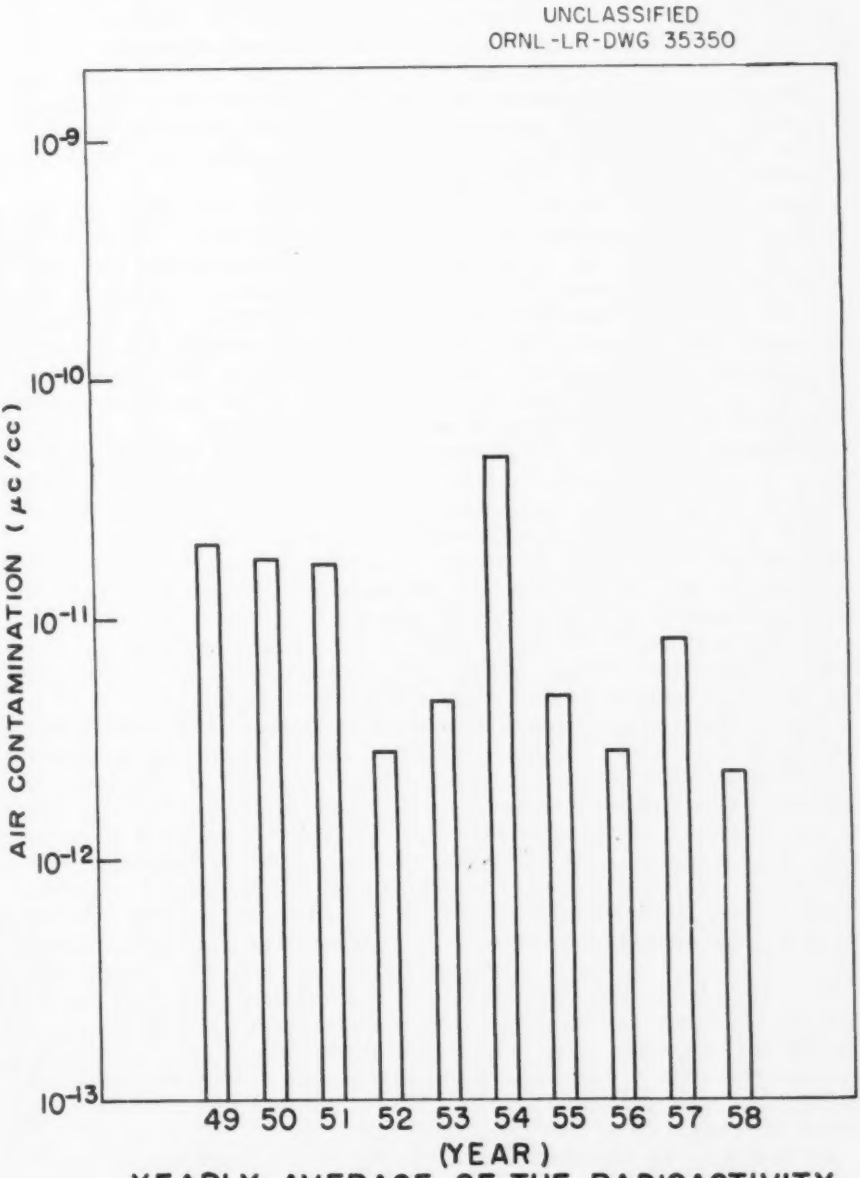
Radioactive atmospheric pollution is controlled as, I just indicated, by monitoring the air as it is discharged from the three stacks in the Laboratory area and by a system of sampling stations throughout the Oak Ridge area. This graph [pointing to fig. 2] indicates the results of these measurements within the laboratory area where the highest levels of air contamination are found. The values given are yearly averages of beta and gamma emitting radioactive contamination which in most cases is considerably less than 1 percent of the maximum permissible concentration values given by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

In addition, fallout from all sources is monitored by means of gum paper collections, fallout trays, analysis of rainwater, background radiation measurements, and by aerial surveys.

Each of the methods of air sampling yields slightly different information relative to the nature of the airborne radioactive material. Since the filter sampling equipment is located inside a louvered enclosure the heavier particles will settle to the ground without being collected on the filter and the sample, therefore, will contain only particles which might be considered small enough to be breathable and deposited in the lower respiratory tract. On the other hand, the trays will collect both the light and the heavy particles. Thus the tray samples furnish some information on the amount of radioactive material that would be inhaled and held up in the upper respiratory tract from which it would be swallowed. The background measurements indicate the radioactivity of the contaminated soil and of the radioactive gas such as the reactor produced Argon 41. Rainwater samples will give information on the soluble as well as the insoluble fraction of the radioactive material.

When the radioactive contamination reaches the point of significance relative to the maximum permissible values recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection, investigations are made as to the source of the contamination and remedial action is taken immediately. On one occasion uranium oxide particles were found to be settling out in the Laboratory area and on still another occasion protoactinium particles were discovered. In each case investigation led to the source of the particles and the processes causing the local fallout were stopped until corrective measures had been completed. The remedial action in these cases required the installation of additional filters in the off-gas system and a program for the routine scanning of the channels in the graphite reactor.

AIR CONTAMINATION ($\mu c/cc$)



YEARLY AVERAGE OF THE RADIOACTIVITY OF AIR IN THE ORNL PLANT AREA

FIG. 2

On some occasions in limited areas of the Laboratory and for short periods of time the air concentration has exceeded the maximum permissible concentrations for continuous occupational exposure. However, immediate action was taken to reduce the air concentration and in no case have the concentrations exceeded 10 percent of the maximum permissible values when averaged over a 13-week period as specified by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

I pause here to emphasize that although for very short periods of time the air contamination in the Laboratory area has exceeded the level for continuous operation, at no time have we exceeded the maximum permissible concentration when averaged over a 13-week period, and the National Committee on Radiation Protection and the International Commission on Radiological Protection specify that you may average over a 13-week interval. In fact, we have not exceeded 10 percent of the occupational exposure values. (See fig. 2 for yearly averages of air contamination in the Laboratory area.)

Radioactive particles which could be attributed to Oak Ridge National Laboratory have been detected at distances up to 40 miles from the area. Although the radioactive material carried by these particles cannot be considered hazardous continuing vigilance is required and constant attention must be given to improving the efficiency of the monitoring stations for detecting radioactive airborne contamination and providing an early warning if high levels are reached at any time.

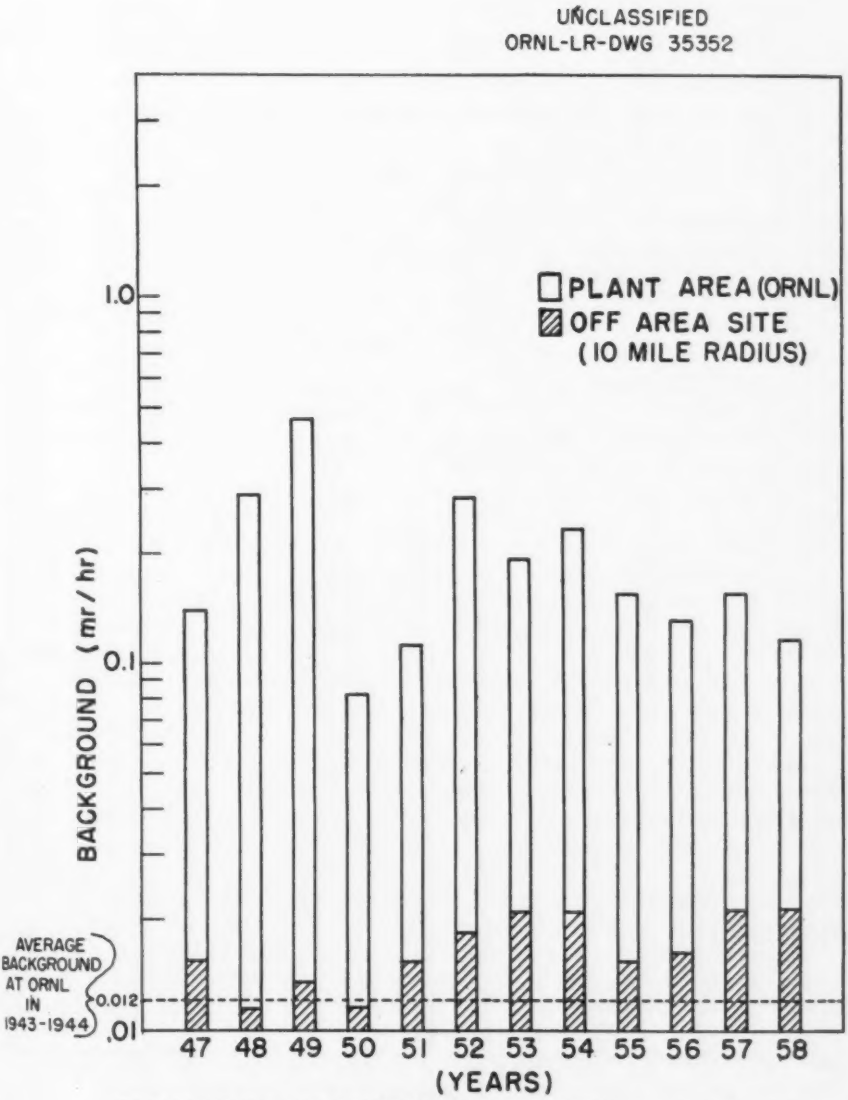
The contamination of the ground in the Laboratory area is monitored by means of routine background measurements at 3 feet above the ground at some 50 locations throughout the Laboratory.

The average level in the laboratory area during 1959 was 0.12 mr per hour (see fig. 3), or about 10 times the average level measured 15 years ago prior to the operation of the Laboratory. However, as shown in figure 3a, this increase in the background in 1958 was only about 4 percent of the maximum permissible occupational exposure rate, as recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection. Nevertheless, increased efforts have been made during the past 10 years to stem the gradual rise in background and we feel encouraged, in spite of the fact that we have increased by several orders of magnitude the levels of work with radioactive material in the Laboratory during this period, there has been no increase in the average background in the Laboratory area. You will note in figure 3a that during the past several years there has been a gradual decrease in the amount of radioactive contamination in the Laboratory in spite of a large increase in operations of the Laboratory.

As indicated by the shaded areas in figure 3a, there has been considerable fluctuation in the background in the neighborhood of the Laboratory.

This in considerable part, has been due to weapons tests in the South Pacific, in Nevada, and in other parts of the world.

I have marked in figure 3a some of the periods of weapons tests, both by our country and by the U.S.S.R. and Great Britain. These weapons tests in considerable measure account for this buildup of radioactivity in the neighborhood of our Laboratory.



BACKGROUND MEASUREMENTS IN ORNL AND VICINITY

FIGURE 3

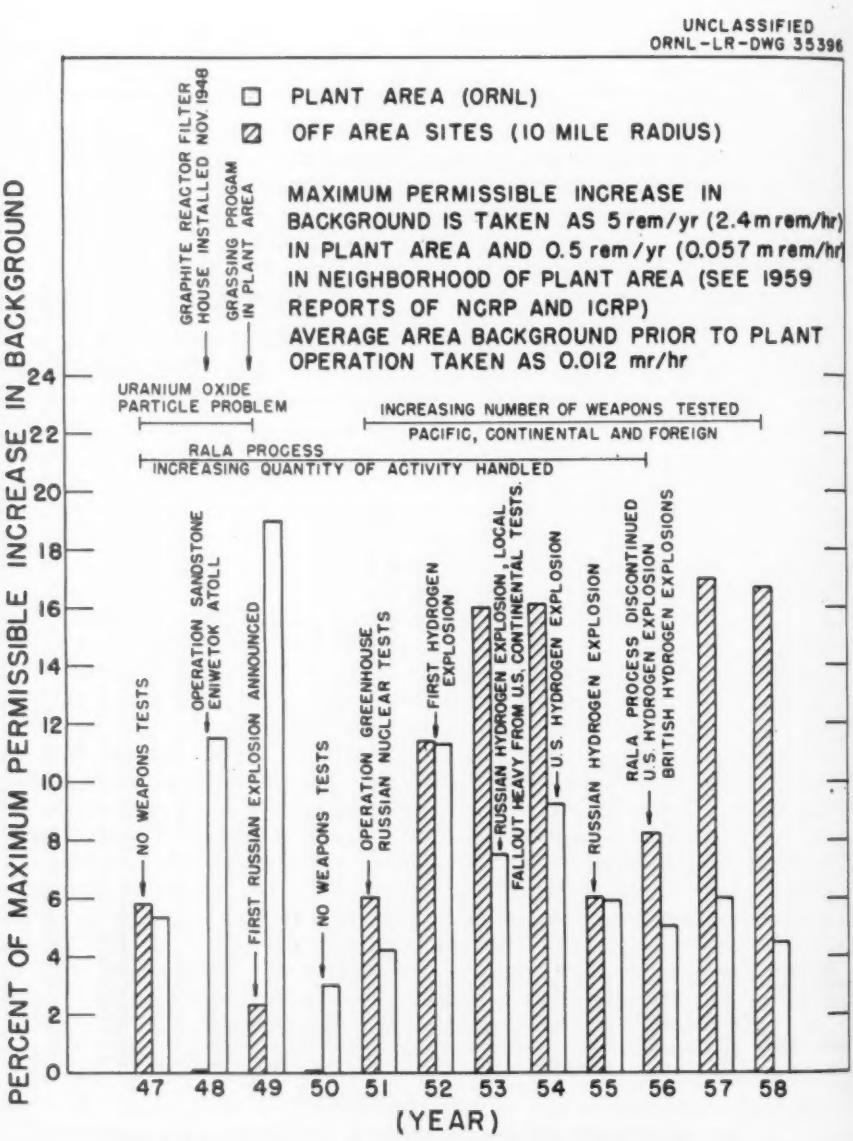


FIG. 3A PERCENT OF MAXIMUM PERMISSIBLE INCREASE IN BACKGROUND IN ORNL AND VICINITY

Notice that there is a change in scale of 10 between the permissible level in the neighborhood of our Laboratory and the permissible level in the Laboratory area. The increase that we measured in background in the neighborhood of the Laboratory during 1957 and 1958 was only about 17 percent of the maximum permissible exposure recommended by the National Committee on Permissible Protection and the International Commission on Radiological Protection. That is, these organizations have recommended that the maximum permissible level for persons living in the neighborhood of such a controlled operation must be kept to not more than 10 percent of the occupational values that we apply inside the Laboratory area, and we have been able to stay below 17 percent of this value.

In the Laboratory area the measurement of underground radioactivity detected by well monitoring reveals the levels to be insignificant. In the waste pit areas where most of the disposable high level radioactive waste is sent radioactivity has been detected in some of the monitoring wells, but it has been established that the contaminant is primarily ruthenium 106 for which the maximum permissible concentration is relatively high. In fact, the maximum permissible concentration for ruthenium 106 is 100 times that of strontium 90 and it is perhaps fortunate that ruthenium 106 therefore serves as a built-in tracer that helps us detect where the more dangerous materials may go. A few surface seeps have developed in the pit area, but the location of the area is such that the direction of ground water and surface water flow is toward White Oak Creek and the Clinch River into which low level liquid wastes are released (as indicated by the earlier discussions) and the water in White Oak Creek and the Clinch River has been monitored since the beginning of the Laboratory operation.

At the beginning of Laboratory operations a small dam was located on White Oak Creek for forming a lake and impounding radioactive waste. This lake served a useful purpose in the early years and removed considerable radioactive material from the water leaving the Laboratory. However, after years of operation the lake became much less effective because the settling of the radioactive waste into the mud produced a state of equilibrium so that as much radioactive material was leaving the lake as was being put into it. There existed the impending possibility of a large release of radioactive material as a result of heavy rains or floods. In addition, a fish population developed and the lake became a favorite stopover point for migratory fowl. To eliminate the possible problems from the sudden failure of the dam and from contaminated fish and migratory fowl, the fish in the lake were killed and the lake was drained.

The dry White Oak Lake Basin has a capacity of 10 million cubic feet. In the event of an accident involving the release of large volumes of radioactive liquids, engineering studies indicate that under normal flow conditions a 23-day delay in the release could be provided by closing the gates at White Oak Dam and once the radioactive material has been confined to form a lake, a bypass could be cut around the lake and dam, thereby providing a longer holdup time. It should be pointed out again that as a supplement to this holdup procedure TVA can release a large amount of dilutent water from Norris Lake which is located 60 miles upstream from the confluence of the White Oak Creek with the Clinch River.

Liquid wastes leaving the Laboratory are sampled at a number of locations. They are sampled routinely at the source where the effluent leaves the Laboratory, at the point where the wastes enter the public waterway (White Oak Creek Dam), and at the nearest center of population downstream from the Laboratory (the K-25 gaseous diffusion plant). Stream gaging operations are carried on continuously at White Oak Creek, Melton Branch, and in the Clinch River to determine creek and river flow in order that dilution factors for evaluating the concentrations of the wastes in the river may be obtained.

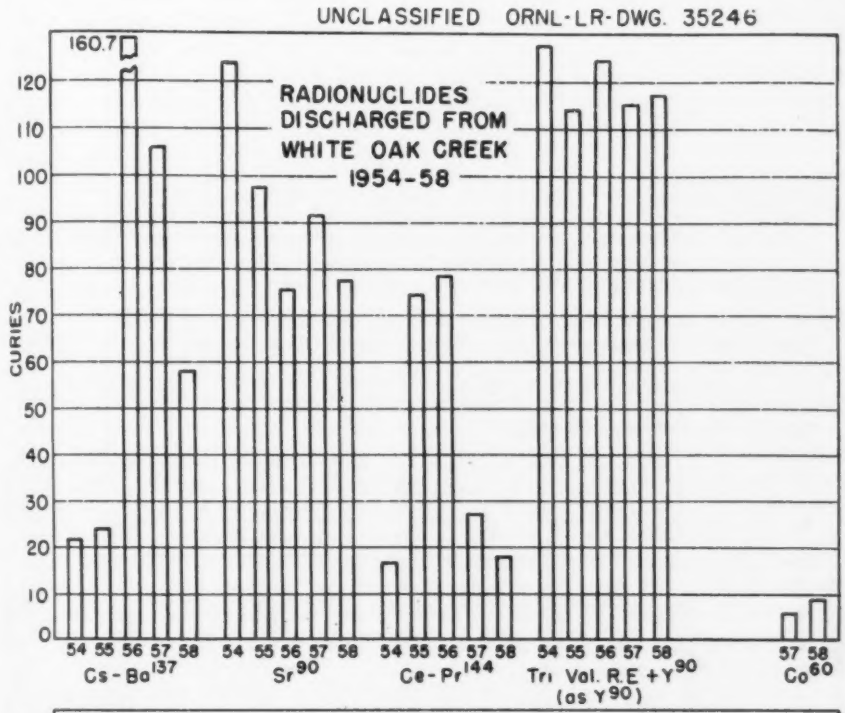
Routine surveys of the Clinch and Tennessee Rivers are conducted to determine the extent of dispersion of radioactive materials in river water and sediment, the levels of contamination encountered and to evaluate the consequent hazards to humans. Periodic measurements and analyses are required to predict the rate of buildup and constantly determine the effectiveness of the liquid management program. Also, information is obtained relative to the effect on future industry of increasing the radioactive content of the water and the botom sediment in the Tennessee River system. Figure 4 gives the results of some of the composite radiochemical analyses, showing the principal radionuclides discharged from White Oak Creek. You will notice we had cesium-barium 137, strontium 90, cesium-praseodymium 144, the rare earths, and cobalt 60 as the principal contaminants discharged into the public water system during these periods.

The lower portion of figure 4 shows the same radionuclides and their buildup in the silt of the Clinch River and the Tennessee River system.

Figure 5 shows the concentration of the radioactive materials in the water of the Clinch River, just below the outfall of White Oak Creek (assuming uniform mixing). These values are compared with 10 percent of the calculated maximum permission values for occupational exposure (values obtained by applying the latest data given in the 1959 revisions of the internal dose reports of the NCRP and the ICRP) as specified for the neighborhood of our operation by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

Although the radioactive contamination in the Clinch River water is less than the calculated values of MPC for each year shown in this figure, the levels in the years prior to 1954 had risen steadily and in fact by 1954 they were only slightly less (i.e., about 50 percent less) than the permissible values. This led to the adoption of more effective radioactive waste control measures such as the discontinuance of the impoundment of contaminated water in White Oak Creek, much extensive use of liquid waste disposal pits and the installation of the process water treatment plant that I described previously.

There are no specific maximum permissible concentration values for the population at large (I refer to the population at large as against the population in the neighborhood of the controlled area). However, the International Commission on Radiological Protection has suggested that for planning purposes an interim average values might be used which are one one-hundredth of the occupational exposure values as applied to the gonads (i.e., for genetic exposure) or one-thirtieth of the values of MPC for occupational exposure to the other body organs (i.e., for somatic exposure). Therefore, the values of one one-hundredth of the calculated maximum permissible concentration values for occupational exposure are given in figure 5 only to


UNCLASSIFIED ORNL-LR-DWG. 35246
160.7
RADIONUCLIDES DISCHARGED FROM WHITE OAK CREEK 1954-58
CURIES
0
10
20
30
40
50
60
70
80
90
100
110
120
54 55 56 57 58
Cs - Ba137
Sr90
Ce - Pr144
Tri Vol. R.E. + Y90 (as Y90)
57 58
Co60

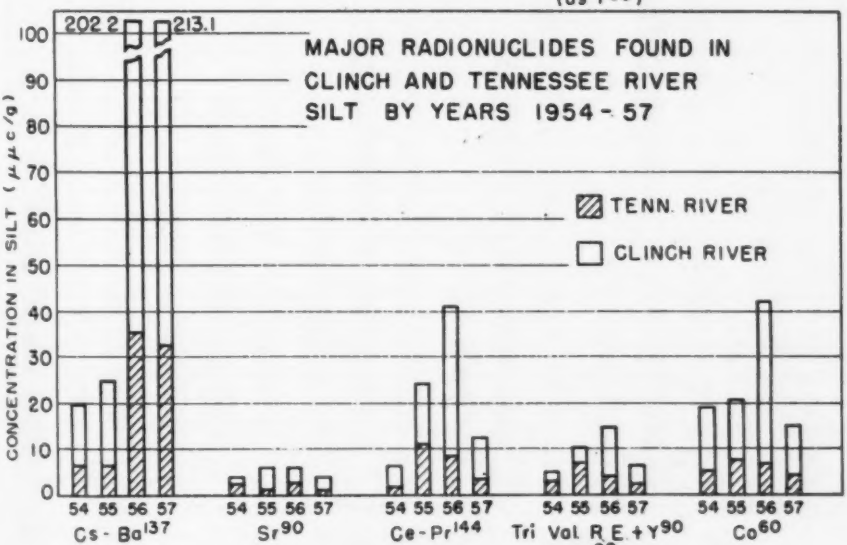

202.2
213.1
MAJOR RADIONUCLIDES FOUND IN CLINCH AND TENNESSEE RIVER SILT BY YEARS 1954 - 57
TENN. RIVER
CLINCH RIVER
CONCENTRATION IN SILT (μμc/g)
0
10
20
30
40
50
60
70
80
90
100
54 55 56 57
Cs - Ba137
Sr90
Ce - Pr144
Tri Vol. R.E. + Y90 (as Y90)
Co60

FIGURE 4

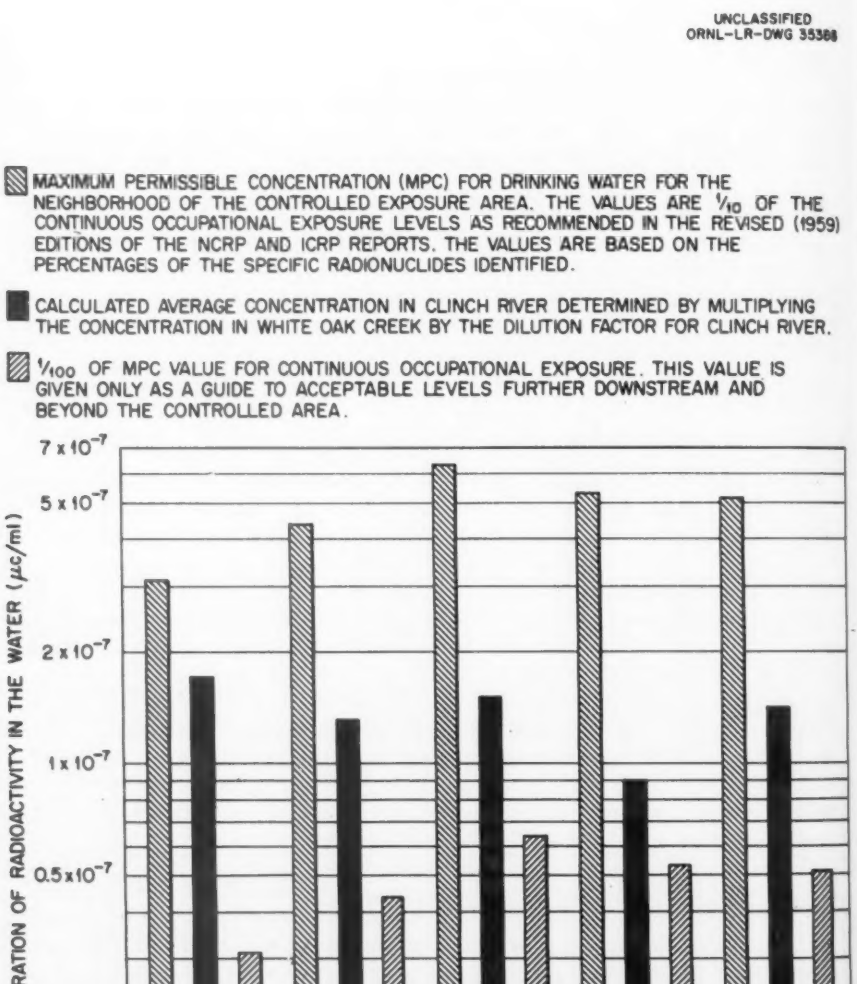

UNCLASSIFIED
ORNL-LR-DWG 35388
MAXIMUM PERMISSIBLE CONCENTRATION (MPC) FOR DRINKING WATER FOR THE NEIGHBORHOOD OF THE CONTROLLED EXPOSURE AREA. THE VALUES ARE 1/10 OF THE CONTINUOUS OCCUPATIONAL EXPOSURE LEVELS AS RECOMMENDED IN THE REVISED (1959) EDITIONS OF THE NCRP AND ICRP REPORTS. THE VALUES ARE BASED ON THE PERCENTAGES OF THE SPECIFIC RADIONUCLIDES IDENTIFIED.
CALCULATED AVERAGE CONCENTRATION IN CLINCH RIVER DETERMINED BY MULTIPLYING THE CONCENTRATION IN WHITE OAK CREEK BY THE DILUTION FACTOR FOR CLINCH RIVER.
1/100 OF MPC VALUE FOR CONTINUOUS OCCUPATIONAL EXPOSURE. THIS VALUE IS GIVEN ONLY AS A GUIDE TO ACCEPTABLE LEVELS FURTHER DOWNSTREAM AND BEYOND THE CONTROLLED AREA.
CONCENTRATION OF RADIOACTIVITY IN THE WATER (μc/ml)
7 x 10⁻⁷
5 x 10⁻⁷
2 x 10⁻⁷
1 x 10⁻⁷
0.5 x 10⁻⁷
0.2 x 10⁻⁷
0.1 x 10⁻⁷
1954
1955
1956
1957
1958
YEAR

Fig. 5. Concentration of Radionuclides in Clinch River (1954–1958) and the Calculated MPC Values.

serve as a guide for future planning of the levels to be maintained farther downstream below the outfall of White Oak Creek and beyond the controlled area. The confluence of the Clinch River with the Emory River and with the Tennessee River and the settling out of the radioisotopes in the river bottom mud provide water concentrations beyond the control area that are less than the values suggested by the International Commission for planning purposes, and in fact, they are of the order of natural background.

Measurements of radioactivity in the water and river bottom sediment and samples of radiochemical analysis are taken every 2 miles in the Clinch River and approximately every 10 miles in the Tennessee River system. These measurements are ordinarily made at a distance of about 100 miles downstream from the Laboratory. On one occasion they were made downstream for a distance of about 590 miles (all the way to the Ohio River). A cross section study consists of making measurements of the radioactivity of the water and river bottom sediment and collecting sediment samples at predetermined intervals along the traverse from one bank to the other. The samples are analyzed radiochemically for the long-lived radioactive isotopes that are present. The radioactive contamination found in the river bottom sediment is usually much higher than found in the river water and reaches a peak (as indicated in figure 6) of about 18 times the average background and this peak is located at a distance of about 12 miles below the outfall of White Oak Creek into the Clinch River. The concentration of the radioactive material in the sediment of the Clinch River drops off materially after the first 20 miles. It is about twice background at 100 miles downstream and approaches background at 150 miles downstream where fission products are still detectable by silt analysis. As shown in figure 7, the level of radioactive contamination in the river sediment drops off suddenly below each dam. It increases again as the next dam is approached and again drops suddenly beyond the dam as the velocity of the water becomes so great that it does not offer time for the sediment to collect on the river bottom.

Samples of water obtained routinely from the Clinch River at the Oak Ridge gaseous diffusion plants and at Kingston, nearest population centers downstream, contain detectable amounts of fission products, but the levels are well below those permitted by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

In conclusion, Oak Ridge National Laboratory is contributing some radioactive material to the local environment and continuing vigilance must be maintained to determine the nature and extent of these radioactive materials. To date, the data from the health physics monitoring systems indicate that the radioactive material released is below the levels recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

This last slide (figure 8) shows graphically and rather forcefully the build up of background radiation as measured in January 1959 in the neighborhood of the laboratory. This is a small map indicating the area in the direction of the prevailing winds from our laboratory. The area extends to Chattanooga which is about 100 miles from the laboratory.

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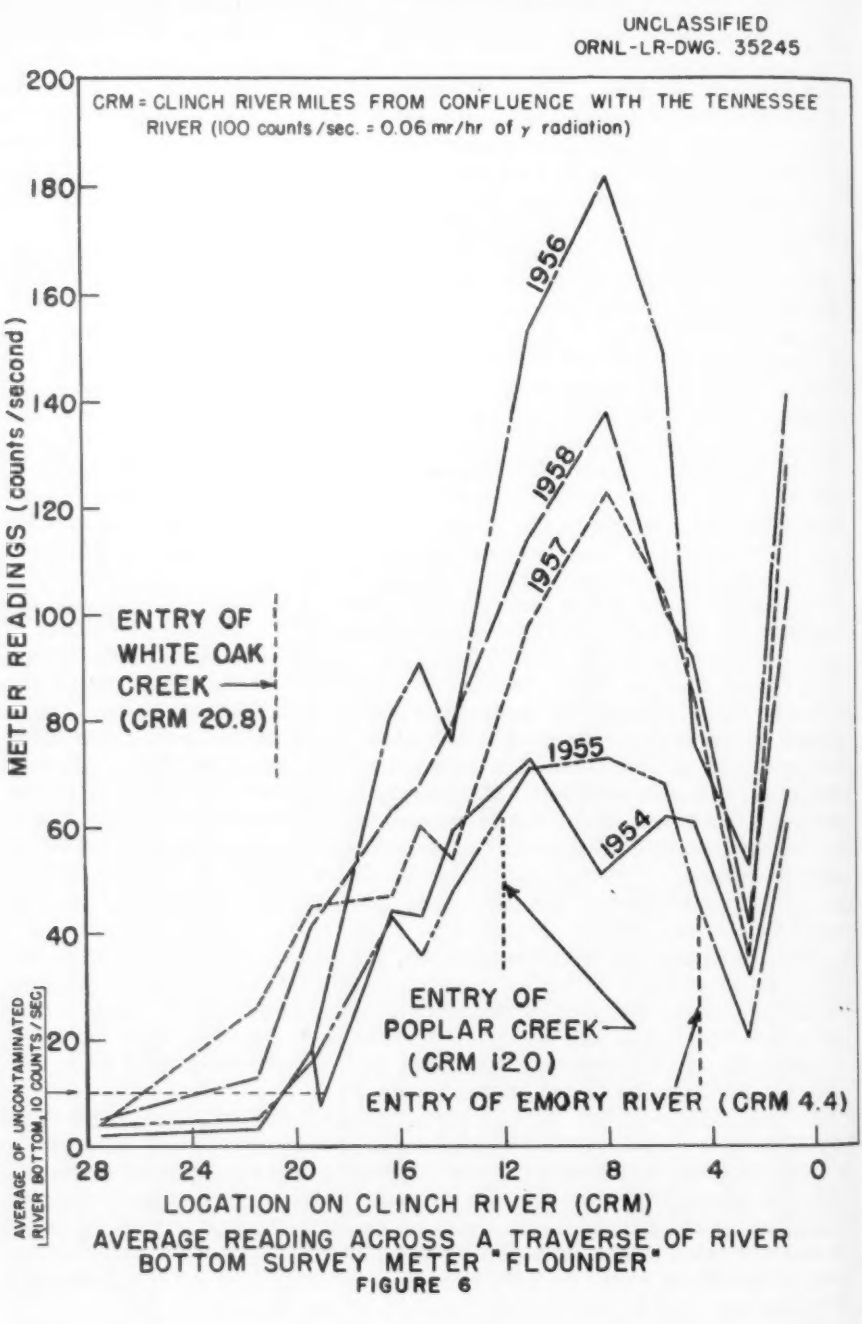
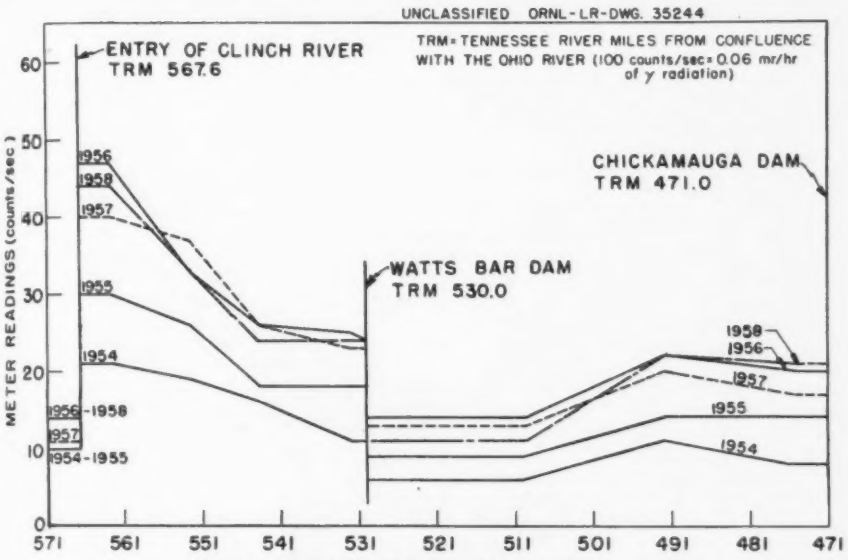

UNCLASSIFIED
ORNL-LR-DWG. 35245
CRM = CLINCH RIVER MILES FROM CONFLUENCE WITH THE TENNESSEE RIVER (100 counts/sec. = 0.06 mr/hr of γ radiation)
METER READINGS (counts/second)
200
180
160
140
120
100
80
60
40
20
0
1956
1958
1957
1955
1954
ENTRY OF WHITE OAK CREEK (CRM 20.8)
ENTRY OF POPLAR CREEK (CRM 12.0)
ENTRY OF EMORY RIVER (CRM 4.4)
AVERAGE OF UNCONTAMINATED RIVER BOTTOM, 10 COUNTS/SEC.
28
24
20
16
12
8
4
0
LOCATION ON CLINCH RIVER (CRM)
AVERAGE READING ACROSS A TRAVERSE OF RIVER BOTTOM SURVEY METER "FLOUNDER"

FIGURE 6



AVERAGE READINGS ACROSS A TRAVERSE OF RIVER BOTTOM SURVEY METER "FLOUNDER"

FIGURE 7

You notice the background in the neighborhood cities and the sharp increase in background in the Laboratory area. However, as indicated in figure 3a this increase in 1958 was only 4 percent of the maximum permissible levels allowable for occupational work. A considerable portion of the increase above the expected average background beyond the laboratory area is due to fall-out from weapons tests. However, some of the activity to the northeast of the laboratory area is due to Laboratory contamination, but as indicated in figure 3a this has not exceeded 17 percent of 10 percent of the occupational value as permitted in the neighborhood of a controlled area.

Thank you.

(The formal statement of Dr. Morgan is as follows:)

WASTE MANAGEMENT PROGRAM AT OAK RIDGE NATIONAL LABORATORY

(By Dr. K. Z. Morgan)

The two primary objectives in waste management are radiation protection and economy. Radiation protection requirements are based upon recommendations of the National Committee on Radiation Protection (NCRP) and the International Commission on Radiological Protection (ICRP). The economics of waste disposal are of primary concern if atomic energy is to compete with present industrial power. Perhaps it should be emphasized that safety and economy are not independent entities for there is no denying that health physics protection is expensive; however, the final cost of an operation can be increased exceedingly if we fail to provide sufficient health physics protection.

There are two basic methods generally in use for the disposal of radioactive waste: (1) the waste may be concentrated and the concentrate stored, or (2)

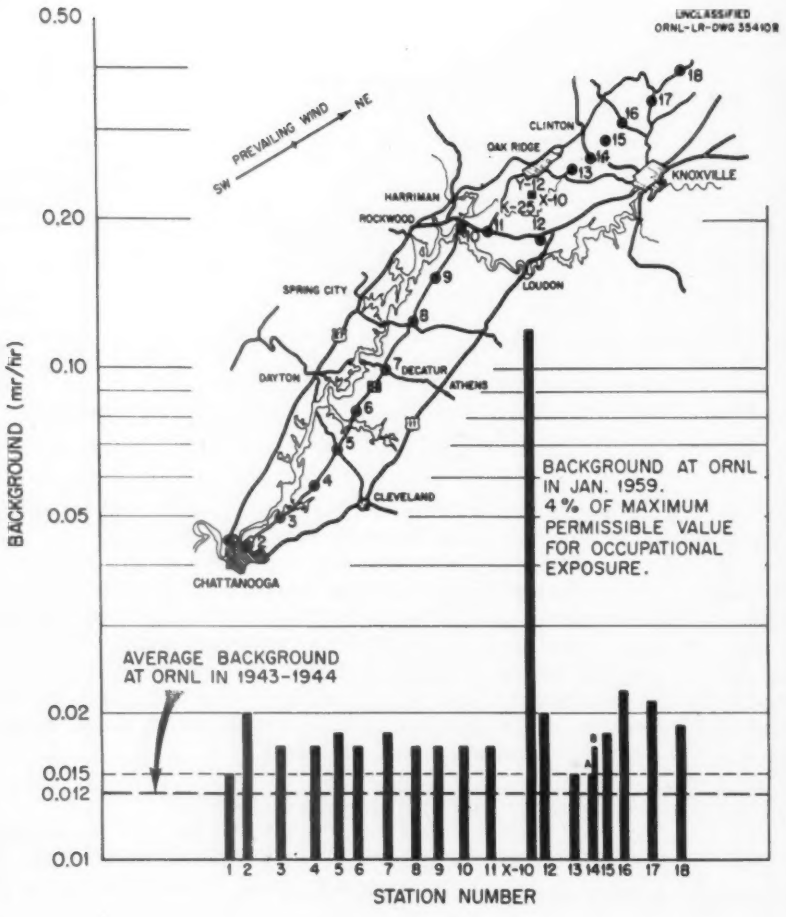


Fig. 8. External Environmental Radiation Measurements at ORNL and Vicinity (Background Measurement mr/hr during Jan. 1959, at 3-feet Above Ground)

the waste may be combined with a passive medium and dispersed throughout the atmosphere, soil, and/or water such that its concentration is reduced to acceptable levels.

Fortunately, the Oak Ridge National Laboratory is located in an area which affords several local advantages as follows: (1) the relative isolation of the Laboratory, (2) the large volume of water available in the Clinch River which partly surrounds the Oak Ridge area, and (3) the extensive Conasauga shale formation. The boundaries of the AEC-controlled area are, at the closest point, about 2 miles from the Laboratory. The nearest large city, Knoxville, Tenn., is approximately 20 miles from the controlled area. Surveys have shown that all the surface and ground water drainage from the Laboratory and waste disposal areas lead to an impoundment fed by White Oak Creek which flows to the Clinch River. The White Oak Creek-Clinch River drainage system has been monitored for radioactive contamination since the beginning of operations at the Laboratory in 1943. Dilution afforded by the Clinch River lowers liquid waste concentration to acceptable levels. The Clinch River flows into the Tennessee River 20.8 miles below the outfall of White Oak Creek, and both rivers form a part of the TVA system of rivers and reservoirs. Thus, in case of an emergency, TVA may be called upon to adjust the flow of the river system to provide holdup and/or dilution. Seepage pits provide a safe and relatively cheap means of disposal for a significant fraction of the liquid and solid radioactive waste from the Laboratory. The Conasauga shale formation allows for seepage pits as the shale has the ability to capture and hold certain of the radioactive components while allowing water to seep through.

WASTE MANAGEMENT FACILITIES [1]

If you will now refer to the flow diagram (figure 1, p. 431) which you have been given, it will assist in following my presentation on Waste Management Facilities. This simplified diagram shows the types of wastes that are handled and the methods used for their disposal. Please note the numbers (1 through 7) near the top of the diagram; I shall refer to each number before describing the portion of the operation listed below that number.

Tracing the flows of material from the top to the bottom of the diagram, it is apparen that Oak Ridge National Laboraory can dispose of its wastes in four ways: (1) by discharge into the atmosphere through tall stacks, (2) by storage and decay, (3) by fixation in the soil of the waste disposal area, and (4) by discharge to the Clinch River. The continuing goal of the Laboratory in its waste disposal program is to immobilize as much as the radioactive material as possible by fixation in the soil or by deposition as insoluble sludge in tanks and pits and to hold to a minimum the amount of radioactive material discharged to the Clinch River and into the atmosphere.

Highly radioactive liquid chemical waste

Reading under No. 1 at the top of the flowsheet, you will note a flow of highly radioactive liquid chemical waste of 7,000 gallons per day. The term "highly radioactive" is used in comparing this waste to our other Laboratory wastes and is not to be confused with the term as used by other installations. The highly radioactive waste at the Laboratory has a concentration of radioisotopes of 1/1000 to 1/100 curies per gallon, while at large production plants the concentration may be several thousand curies per gallon. I mention this not to minimize our waste disposal problems—since we are aware that even small amounts of some of the radioisotopes when improperly handled can be extremely hazardous—but to give you some yardstick for comparing our disposal problems to those of the production plants.

Most of the highly radioactive waste is produced in chemical processing buildings and chemistry laboratories. It is discharged from laboratory sinks and process drains through stainless steel pipes into 19 underground, stainless steel, monitoring tanks that are located near the waste-producing buildings. The rate of waste accumulation at all 19 monitoring tanks is continuously recorded at one

[1] This section has been abstracted from that part of the written record prepared by F. N. Browder et al., for submission to this hearing.

location through a telemetering system, and to ensure that none of the tanks will overflow in case the telemetering system fails, the tank volumes are manually checked every 4 hours.

When a monitoring tank fills, the waste is pumped to three underground concrete central storage tanks that have 170,000 gallons capacity each. The material is temporarily stored in these tanks to allow the short-lived radioisotopes to decay; it is then pumped to waste pits located about 1½ miles from the Laboratory. The waste pits are three 1 million gallon capacity open pits excavated in the earth in a location chosen for its remoteness from the Laboratory, the type of soil, and the natural underground water drainage to White Oak Creek. This particular soil, made up of clay and Conasauga shale, removes and retains most of the radioisotopes while the solutions seep slowly through it toward White Oak Creek. The waste discharged into the pits is sampled and analyzed for radioactivity, and its movement in the soil and drainage into the Creek is monitored periodically by health physicists.

Liquid uranium waste

The liquid uranium waste flow (shown under No. 2 of the flowsheet) is also a product of the chemical processing operations and research. This type of waste is presently produced in very small quantities averaging no more than 100 gallons per week and is not a true waste in the strict sense of the word, since it is a solution of reusable uranium contaminated with fission products. This material is collected and stored in underground stainless steel monitoring tanks (seven in number) where a check is kept on the sources of wastes, and the volumes in all tanks are determined by manual checks and by a continuously recording telemetering system. When a tank is filled, the waste is pumped into 42,500 and 170,000 gallon underground concrete central storage tanks where it is held until transferred to a small processing plant for uranium recovery. The radioactive products separated from the uranium are transferred in solution to the central storage tanks for highly radioactive chemical waste while the decontaminated uranium is returned to regular channels for reuse.

Process waste water

The daily 700,000 gallons of mildly contaminated process waste water (listed under No. 3) is the least radioactive of all Laboratory wastes; yet it is the most difficult to manage because of its large volume. This waste is made up mainly of water used for cooling process equipment and water from floor drains, and under normal conditions carries a radioactive concentration of only one-ten thousandth of that handled by the highly radioactive chemical waste system first described. A considerable amount of the radioactive liquid put through this system is the result of equipment failure, human error, or accidents which cause a misdirection of the radioactive waste from the system for the highly radioactive material. Whenever these incidents take place an immediate effort is made to correct the conditions in order to direct the radioactive material to the high level system.

Because of the large quantity of process waste generated per day, the storage of it for any appreciable time is impractical. This situation leaves the Laboratory practically no choice other than to dispose of it on a current basis into White Oak Creek. At one time, when the levels of radioactive material handled by the Laboratory processes were relatively very low, the procedure was simple; the low level waste was discharged to the creek without treatment and comparatively little effort was required in monitoring the waste streams before discharge to the creek. During the last 13 years, as the Laboratory expanded and the levels of processed radioactive material increased by a factor of about 100, methods had to be developed for monitoring all the waste streams and for removing most of the radioisotopes from the waste before it could be discharged safely to the creek.

Control over the discharge to the creek is accomplished mainly by continuously checking the underground network, which resembles a sanitary sewer system. This is done by frequently sampling and measuring the various waste streams and their combined waste flow before it passes through the automatic diversion valve for processing or for discharge to the creek. If the analysis of the combined waste indicates a substantial increase in radioactivity over normal conditions, samples taken out of the individual streams are then analyzed

for activity to determine the source. Sometimes personnel are not aware that their operations are generating excessive radioactive waste and thus it is necessary to contact them and make appropriate corrections in their operations.

All the waste generated during the regular working hours is collected and treated before discharge to the creek. During the off-shift hours, however, the waste containing less than 50 counts/min/ml (β,γ) is diverted to the creek through a settling basin, which discharges directly to the creek.

From 53 percent to 87 percent of the total radioactivity contained in the waste diverted into the process waste treatment plant is removed with a single pass through the plant; 73 percent to 87 percent of the Sr^{90}, considered to be the most hazardous component, is removed. A higher percentage of removal can be expected by repeating the treatment.

The process employed in this plant for the treatment of radioactive wastes is new in atomic energy operations, it is the first plant built that utilizes on a large scale a standard but modified lime-soda ash process normally used for water softening. The plant has been in operation for only about a year and many variations of the process remain to be tested; therefore its full efficiency may not yet be realized. It is probable that this plant will serve as a model for other atomic energy installations having the problem of disposal of large volumes of low level radioactive wastes.

Solid waste

The disposal of solid waste is shown under No. 4. Practically all the solid waste produced by the Laboratory consists of paper, glassware, and other pieces of laboratory equipment, much of which is contaminated with radioactive materials. Occasionally building materials such as concrete and wood that cannot be salvaged economically by any known decontamination method must be handled. As compared to the liquid wastes previously described, the solid wastes usually contain an almost insignificant quantity of radioactive material, probably about 1 percent of the total disposed of by the Laboratory. Accurate determination of the quantity of radioisotopes in the solid waste is difficult because the waste is composed of a nonuniform mixture of many materials.

In addition to being used for disposal of Laboratory wastes, the burial grounds are used for disposal of wastes from other AEC installations and contractors and the Laboratory's radioisotope customers. Radioisotope customers contribute a comparatively insignificant volume; other installations, mainly Argonne National Laboratory, Knolls Atomic Power Laboratory, Mound Laboratories, Battelle Memorial Institute, and General Electric in Evansdale, Ohio, account roughly for half the land area assigned permanently to this burial program. To date, 25 acres have been used by the burial ground operation. The current rate of use is approximately 5 acres per year.

Waste gases

Before reviewing the waste gas disposal operations, it is necessary to give some explanation of the term "waste gas." Actually, a more descriptive term would be "contaminated air" since the bulk of the waste is composed of air that is contaminated with radioactive materials. The radioactive contaminants may be either gases or microscopic solid particles—usually a combination of both.

Because of extremely large volumes, it is impractical to store most gaseous wastes for any appreciable period of time. The only practical methods of disposal must, therefore, involve continuous decontamination of the gases as they are produced and discharged to the atmosphere.

The 120,000 cubic feet per minute of air used for cooling the graphite reactor, as shown under No. 5, is drawn through the reactor by means of two fans operating at a capacity of 60,000 cubic feet per minute. After leaving the reactor and before going through the fans, the air is put through a set of filters that remove the radioactive particulates which, if discharged into the atmosphere, would create the principal hazard from airborne radioactive contamination. The air, containing radioactive gases, is discharged into the atmosphere through a stack 200 feet high. The filters holding the radioactive particulate matter are periodically removed and buried in the waste disposal area.

The cell and hood ventilation system is a network of ducts that collects under a low vacuum a relatively large volume of exhaust air from laboratory hoods and concrete shielded cells used for processing radioactive materials. This system is

illustrated under No. 6 on the diagram. The air is drawn through the system by 5 electrically driven fans having capacities ranging from 25,000 to 60,000 cubic feet per minute; in case of failure of the electrical equipment, three steam driven fans are used. The air from this system is discharged into the atmosphere through 2 stacks, one 200 feet and the other 250 feet high.

The chemical processing of radioactive materials usually results in a release of highly concentrated radioactive contaminants in a relatively small volume of gas. These gases are handled by the highly radioactive gas (or off-gas) system shown under No. 7. The gases are collected and drawn by vacuum through a network of underground stainless steel lines and are discharged, after cleanup, through the stack 250 feet high, used for the discharge of some of the cell ventilation air previously described.

The system just described is the primary radioactive gas disposal facility, while the cell ventilation system is the secondary system for all large scale chemical processing. The off-gas system is connected directly to the processing vessels, whereas the cell ventilation system is connected only to the cell where it removes the air from the area in which the vessels are located. Unless there is a failure all the gaseous activity is removed by the off-gas system, keeping the air drawn into the cell ventilation system free of contamination. The cell ventilation system is used, therefore, as a secondary standby for preventing the spread of contamination in the event of a leak from the off-gas system. Keeping the off-gas system operationally reliable and using it properly in the processing cells so that the cell ventilation air does not become contaminated reduces the volume of gas that must be decontaminated and reduces the total radioactive material that is discharged to the filters and through the stacks.

The decontamination of the gases handled by the off-gas system is accomplished partially in the operating cells before the gases are exhausted into the system. The bulk of the chemically reactive gas is removed in the operating cells by scrubbing with caustic. The solid particles and microscopic droplets of liquid carried by the gas stream are removed by passage through an electrostatic precipitator and by filtration before the gases are exhausted into the stack. The radioactive material collected in the precipitator is periodically washed out with water that drains into the liquid waste system. The filters are disposed of in the burial ground.

THE EFFECTIVENESS OF WASTE MANAGEMENT AT OAK RIDGE NATIONAL LABORATORY (ORNL)

Let us now turn our attention to the effectiveness of waste management at ORNL. The method for determining effectiveness involves a series of monitoring techniques which will be described next.

Air monitoring

Radioactive atmospheric pollution is monitored and controlled by the following methods: (1) Air discharged from the three stacks in the Oak Ridge area is monitored continually at the stacks for the purpose of preventing excessive discharges of radioactive contaminants to the atmosphere; (2) a system of sampling stations using air filtering techniques is maintained throughout the Oak Ridge area, quite apart from stack monitoring, which allows for continuous evaluation of the radioactivity in the air. Figure 2 (p. 435) indicates the results of these measurements made within the Laboratory operating area where the highest levels of air contamination are found. The values given are the yearly averages (of the β and γ emitting radioactive contamination) which in most cases are considerably less than 1 percent of the MPC values given by the NCRP and the ICRP for this type of occupational exposure; (3) fallout from all sources is monitored by means of gum paper collectors, fallout trays, and analysis of rainwater samples that are collected; and (4) background radiation measurements are made periodically at fixed points on the ground and by aerial surveys to determine whether or not there is significant fluctuation in the background radiation.

Each of the methods of air sampling yields slightly different information relative to the nature of airborne radioactive material. Since the filter sampling equipment is located inside a louvered enclosure, the heavier particles will settle to the ground without being collected on the filter and the sample

will contain only breathable particles which might be considered small enough to become deposited in the lower portions of the respiratory tract. The fallout trays will collect the heavier particles as well as the light particles, and thus these tray samples represent the total fallout and furnish some information on the amount of radioactive material that would be inhaled and held up in the upper portions of the respiratory tract from which it would be swallowed and passed into the gastrointestinal tract. The background measurements indicate the radioactivity of the contaminated soil and of radioactive gas such as the natural radon or the reactor-produced argon 41. Rainwater samples will contain both fallout and "rainout" and will give information of the soluble as well as the insoluble fractions of the radioactive material.

From these montitoring samples, we are able to determine the general level of air and soil contamination in the area, the source of particular contamination problems, and the effectiveness of various additional radiation control measures that have been applied during the past few years. When the radioactive contamination reaches a point of significance relative to the maximum permissible values recommended by the NCRP and ICRP, investigations are made as to the source of the contamination and remedial action is taken immediately. On one occasion uranium oxide particles were found to be settling out on the Laboratory area. On still another occasion protoactinium particles were discovered. In each case investigations led to the source of the particles, and the processes causing the local fallout were stopped and remained inactive until corrective measures had been completed. The remedial action in the above cases required a number of changes, among the more important of which were the installation of additional filters in the off-gas system and a program for the routine scanning of channels in the graphite reactor for defective fuel elements. The roofs and roadways in the area were washed, and the barren areas within the Laboratory area were sown with grass to keep down the radioactive dust particles.

On some occasions in limited areas of the Laboratory and for short periods of time the air concentration has exceeded the MPC for continuous occupational exposure. However, immediate action was taken to bring the values down to acceptable values, and in no case have the concentrations, when averaged over a 13-week period as specified by the NCRP and the ICRP exceeded 10 percent of the maximum permissible values. The 13-week interval is given by the NCRP and the ICRP as the longest period over which the MPC values for occupational exposure may be averaged.

Radioactive particles which could be attributed to the Oak Ridge National Laboratory have been detected at distances up to 40 miles from the area. Although the radioactive material carried by these particles cannot be considered hazardous, continuing vigilance is required and constant attention must be given to improving the efficiency of the monitoring stations for detecting airborne radioactive contamination and providing an early warning if high levels should be reached.

Background measurements

Contamination of the ground in the Laboratory area is monitored by means of routine background measurements at 3 feet above the ground at some 50 locations throughout this area. Figure 3 p. 437) indicates that the average level in the Laboratory area during 1958 was 0.12 mr/hr or about 10 times the average level measured 15 years ago prior to the operation of the Laboratory. However, as shown by figure 3a (p. 438) this increase in the background is only 4 percent of the maximum permissible occupational exposure rate recommended by the NCRP and the ICRP. Nevertheless, increased efforts have been made during the past 10 years to stem the gradual rise in background in the plant and neighboring areas of Oak Ridge and we feel encouraged that in spite of the fact that we have increased by several orders of magnitude the level of work with radioactive materials during these 10 years there has been no increase in the average background in the Laboratory area and the background in the neighboring areas seems to be leveling off at an average of about 0.022 mr/hr, which is about twice the original background level. This increase in background in the neighborhood of our Laboratory is about 17 percent of the maximum permissible exposure recommended by the NCRP and the ICRP for people living in the neighborhood of such an installation.

Well monitoring

Evaluation of the effect of liquid waste disposal with regard to the underground water table in the Laboratory area and in the area of the waste pit operation is made by means of monitoring wells. In the Laboratory area, the measurement of underground radioactivity detected by well monitoring reveals the levels to be insignificant. In the waste pit area where most of the disposable highly radioactive waste materials are sent, radioactivity has been detected in some of the monitoring wells, but it has been established that the contaminant is primarily ruthenium 106 for which the maximum permissible concentration, MPC, in water is relatively high, e.g., the MPC for ruthenium 106 is a hundred times larger than the MPC for strontium 90. A few surface seeps have developed in the pit area, but the location of the area is such that the direction of ground water and surface water flow is toward White Oak Creek and the Clinch River into which low-level liquid wastes are released (as indicated by the earlier discussions) and the water in White Oak Creek and the Clinch River has been monitored since the beginning of the Laboratory operation.

Liquid waste monitoring

At the beginning of the Laboratory operation, a small dam was located on White Oak Creek to create a lake for impounding radioactive wastes to permit settling and decay of the radioactive material. This lake served a useful purpose in the early years and removed considerable radioactive material from the water leaving the Laboratory. However, after years of operation the lake became much less effective because the settling of radioactive waste into the mud produced a state of equilibrium so that as much radioactive material was leaving the lake as was being put into it. There existed the impending possibility of a large release of radioactive material as a result of heavy rains or floods. In addition, a fish population developed and the lake became a favorite stopover point for migratory fowl. To eliminate possible problems from the sudden failure of the dam or from contaminated fish and migratory fowl, the fish in the lake were killed and the lake was drained.

The dry White Oak Lake basin has a capacity of 10 million cubic feet. In the event of an accident involving the release of large volumes of radioactive liquids, engineering studies indicate that under normal flow conditions a 23-day delay in the release could be provided by closing the gates at White Oak Dam and, once the radioactive material has been confined to form a lake, a by-pass could be cut around the lake and dam, thereby providing a longer holdup time. It should be pointed out again that as a supplement to this holdup procedure TVA can release a large amount of dilutent water from Norris Lake which is located 60 miles upstream from the confluence of the White Oak Creek with the Clinch River.

Liquid wastes leaving the Laboratory are sampled at a number of locations. They are sampled routinely at the source where the effluent leaves the Laboratory, at the point where the wastes enter the public waterway (White Oak Creek Dam), and at the nearest center of population downstream from the Laboratory (the K–25 gaseous diffusion plant). Stream gaging operations are carried on continuously at White Oak Creek, Melton Branch, and in the Clinch River to determine creek and river flow in order that dilution factors for evaluating the concentrations of the wastes in the river may be obtained.

Clinch River monitoring

Routine surveys of the Clinch and Tennessee Rivers are conducted to determine the extent of dispersion of radioactive materials in river water and sediment, the levels of contamination encountered, and to evaluate the consequent hazards to humans. Periodic measurements and analyses are required to predict the rate of buildup and, consequently, determine the effectiveness of the liquid waste management program. Also, information is obtained relative to the effect on future industry of increasing the radioactive content of the water and bottom sediment in the Tennessee River system. Figure 4 (p. 441) gives results of some of the composite radiochemical analyses showing the principal radionuclides discharged from White Oak Creek and the buildup of these radionuclides on the silt in the Clinch and Tennessee Rivers during the past few years. Figure 5 (p. 442) shows the concentration of the radioactive material in the water of the Clinch River just below the outfall of White Oak Creek (assuming uniform mixing). These values are compared with the values of

MPC that are applicable in the neighborhood of our operations (i.e., they are compared with 10 percent of the MPC for continuous occupational exposure). The calculated values were obtained by applying the latest data given in the 1959 revisions of the Internal Dose Reports of the ICRP and the NCRP. Although the levels of radioactive contamination in the Clinch River water are less than the calculated values of MPC for each year shown in figure 5, the levels in years prior to 1954 had risen steadily and in fact the level of 1.7×10^{-7} μc/cc reached in 1954 was only slightly less than the calculated MPC of 3.1×10^{-7} μc/cc. This led to the adoption of more effective radioactive waste control measures (e.g., discontinuance of the impoundment of contaminated water in White Oak Lake in October of 1955; more extensive use of the liquid waste disposal pits; installation of the process water treatment plant in August of 1957, etc.).

No specific values of MPC applicable to the population at large have been given by the NCRP and the ICRP, but the ICRP has suggested that for planning purposes ad interim average values might be used which are 1/100 of the values of MPC for occupational exposure to the gonads (i.e., for genetic exposure) or 1/30 of the values of MPC for occupational exposure to the other body organs (i.e., for somatic exposure). Therefore, the values of 1/100 of the calculated MPC for occupational exposure are given in figure 5 only to serve as a guide for future planning of the levels to be maintained further downstream below the outfall of White Oak Creek and beyond the controlled area. The confluence of the Clinch River with Emory River and the Tennessee River and the silting out of the radioisotopes into the river bottom mud provide water concentrations beyond the controlled area that are less than the values suggested by the ICRP for planning purposes (i.e., they are less than 1/100 of the occupational values) and in fact they are of the order of natural background.

Cross-section measurements are taken every 2 miles in the Clinch River and approximately every 10 miles in the Tennessee River and its reservoirs. These measurements are ordinarily made to a distance of about 100 miles downstream from the Laboratory and on one occasion they were made downstream for a distance of about 590 miles (all the way to the Ohio River). A cross-section study consists of making measurements of the radioactivity of the water and river bottom sediment and collecting sediment samples at predetermined intervals along the traverse from one bank to the other. The samples are analyzed radiochemically for the long-lived radioactive isotopes that are present. The radioactive contamination found in river bottom sediment is usually much higher than that found in the river water. The highest average radioactive contamination of river bottom sediment (resulting in an average dose rate of about 0.1 mr/hr on the river bottom) is about 18 times background and as shown by figure 6 (p. 444) is located about 12 miles below the outfall of White Oak Creek into the Clinch River. The concentration of radioactive material in the sediment of the Clinch River drops off materially after the first 20 miles; it is about twice background at 100 miles and approaches background at 150 miles downstream where fission products are still detectable by silt analysis. As shown by figure 7 (p. 445) the level of radioactive contamination in the river sediment drops off suddenly below each dam in the Tennessee River, increases again as the next dam is approached, and again drops suddenly beyond the dam where the water velocity is too great for much settling of the mud to take place.

Samples of water from the Clinch River at the Oak Ridge gaseous diffusion plant and at Kingston, Tenn., the nearest population centers downstream, have indicated detectable amounts of fission products, but levels well below those permitted by the NCRP and the ICRP for water used for continuous drinking. In fact, the drinking water for the employees at the gaseous diffusion plant is obtained from the river at a point approximately 8 miles below the outfall of White Oak Creek.

Conclusion

In conclusion, Oak Ridge National Laboratory is contributing some radioactive materials to the local environment and continuing vigilance must be maintained to determine the nature and extent of these radioactive materials. To date, the data from the health physics monitoring system presently in use indicate that the radioactive material released is below the levels recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection.

Representative HOLIFIELD. Dr. Morgan, your conclusions and your data which you have presented give assurance to this committee that the safety factor is being looked at very carefully in the Oak Ridge operation and it is comforting to know that you are taking measures to see that the environment is not contaminated.

Are there any questions of Dr. Morgan by members of the committee?

Chairman DURHAM. Mr. Chairman, since Dr. Morgan's waste management operations will probably be used as a model in the future, turning back to your page 1, could you give us an estimate as to the cost of such an operation? Has such a determination been made?

I am just thinking out loud here as to the cost of building such plants.

Dr. MORGAN. Mr. Durham, I don't have with me the operating cost figures. If you will permit, Mr. Struxness, who is in direct charge of this program, and who is operating the slide for me, can answer the question now, or when he appears later on the program.

Representative HOLIFIELD. Do you have those answers at this time, Dr. Struxness?

Dr. STRUXNESS. I will have to do a little searching for them.

Representative HOLIFIELD. Are you appearing later?

Dr. STRUXNESS. Yes, sir.

Representative HOLIFIELD. You are appearing tomorrow, I believe?

Dr. STRUXNESS. Yes.

Chairman DURHAM. At the present time is this plant operating at full capacity. In other words, could it take on more contamination from other places if we had to use such an operation?

Dr. MORGAN. This particular plant that I spoke of is new and is not operating at full capacity. It is a lime soda treatment plant similar to the type that is very common in city treatment plants. Its unique feature is that it can operate on very large volumes of radioactive wastes.

Chairman DURHAM. All types?

Dr. MORGAN. All types of radioactive wastes and it is particularly useful in that it removes most of the strontium and cesium, the most dangerous materials. It would not apply to gaseous waste.

Chairman DURHAM. How about the solid wastes?

Dr. MORGAN. The solid wastes are all buried in our burial ground, located a mile and a half from the laboratory in a special shale formation, the Conasauga shale.

Chairman DURHAM. In other words, in this operation at the present time you are using three types of decontamination?

Dr. MORGAN. That is right.

Chairman DURHAM. The gaseous diffusion, the solid wastes, and also the liquid wastes, itself?

Dr. MORGAN. Correct.

Chairman DURHAM. Each one requires a different operation; is that correct?

Dr. MORGAN. That is correct. In the case of the different wastes, there are different levels that require modifications in the procedures.

You asked about the cost. I could give you some idea of this cost. The total cost of our monitoring program (that is, the cost of the entire applied health physics monitoring program at our Laboratory) is 0.16 percent of the Laboratory operating cost. Sixteen hundredths of 1 percent of the Laboratory budget is what it costs the Laboratory to carry out this entire applied health physics monitoring program.

Chairman DURHAM. Then it is relatively low compared to the full operation?

Dr. MORGAN. The ORNL budget runs in the order of $52 million. This (0.16 percent) is quite a bit of money, but percentagewise it is a small investment.

Representative HOLIFIELD. This includes taking care of materials shipped to Oak Ridge?

Dr. MORGAN. That is correct. It does not include our health physics research program.

Chairman DURHAM. Of course, the capital cost is quite a sizable sum?

Dr. MORGAN. Yes, indeed.

Representative HOLIFIELD. Dr. Morgan, I was impressed yesterday with Dr. Wolman's testimony on the point that the recovery of radioisotopes waste products did not constitute the removal of waste disposal problem, you still have these radioisotopes.

Now, we have had in the newspapers recently a prominent use of this as a power source for this satellite battery, so-called satellite battery, and I understand that strontium 90 and cesium 144 are being considered for such use.

As an expert in evaluating the internal hazards presented by radioactive material, how do you view the use of 10,000 curie quantities of strontium 90 in these power sources, power batteries? Is it an important amount from the standpoint of accidental contamination of the environment? Is there something that is a substitute for it which could be used which might produce the same amount of power, but do it more safely?

These are some of the things I would like to have your opinion on.

Dr. MORGAN. As a health physicist, my first and primary interest in life is the protection of personnel from damage from ionizing radiation. As a health physicist, however, one of my purposes in life is to enable man to live safely with atomic energy and to enable atomic energy to compete with other sources.

With these two conflicting objectives, I think my answer would be that we must find all the peacetime applications of fission products that we can. We must move slowly in the application to such devices that might readily contaminate the earth and might become available (for exposure) to man.

So my answer then is that I would consider that radioisotopes are uniquely adaptable to rocket ships because they do not require oxygen for their operation, but they would represent a considerable hazard, an extreme hazard should one of these devices containing strontium fall under certain circumstances where it would not be completely consumed in the atmosphere.

Representative HOLIFIELD. I am interested in the experiments which are going to take place in California, among other places, we are going to be firing the Atlas. If an accidental misfire should occur, is it not true that if it had this material and it should fall on the farmland of California, that there would be a considerable contamination?

Dr. MORGAN. Yes; that is very true. If there would be a misfire of one of these devices such that it would not be completely consumed in the atmosphere and would burn locally in a small region, it might spread considerable contamination in that region and would require emergency measures to see that the personnel were not exposed.

Representative HOLIFIELD. So if there are substitutes for this extremely toxic material which could be used, why, we should move toward those substitutes rather than take chances?

Dr. MORGAN. We certainly should. But the characteristic you are seeking are mutually exclusive in that you want a radionuclide with a large amount of energy, with a long half life; a radionuclide that dissipates most of its energy in the form of alpha and beta particles which are long-lived and these features comprise the features which make it most hazardous to man.

Representative HOLIFIELD. We have to evaluate the use of it then in relation to our military objectives and our scientific objectives of obtaining energy from these battery operated mechanisms in space vehicles?

Dr. MORGAN. That is correct. If we could be sure that they would always leave the atmosphere and acquire enough velocity to be consumed before reentry, then I don't believe there would be much of a serious problem.

Representative HOLIFIELD. As we develop the reliability of missiles, then, we achieve, you might say, a safer position in the utilization of these toxic materials because the chance of misfire is less? Have you ever been consulted on this matter by the Defense Department or the Atomic Energy Commission?

Dr. MORGAN. No; I have not.

Representative HOLIFIELD. Are there any further questions of Dr. Morgan?

Senator HICKENLOOPER. Yes, Mr. Chairman. Dr. Morgan, in regard to the situation just posed by Congressman Holifield, the same would be true about an atomic reactor, if you had a meltdown or some so-called atomic incident which would cause some sort of a burst, there would be contamination of the surrounding area, would there not?

Dr. MORGAN. Yes, sir.

Senator HICKENLOOPER. So we are really dealing with atomic materials here most of which have definite dangerous potentials if they get out of hand for one reason or another by nuclear explosion, or meltdown, something like that, in an atomic plant.

Atomic materials, once they have become active, are pretty dangerous regardless of the form they are in.

Dr. MORGAN. Yes, indeed.

Senator HICKENLOOPER. That is if they come in contact with people or areas in sufficient quantities. With reference to your statement on page 12 of this report I have here, I don't know that I fully understand exactly what you meant in your discussion of the area contamination rate which you indicated had occurred about the time of atomic tests at various places. You did not mean to say, did you, that that increase was peculiar only to the Oak Ridge area?

Dr. MORGAN. No; you would find corresponding increases in other areas.

Senator HICKENLOOPER. I thought that is what you meant. But I did not know whether I fully understood it, or not.

Dr. MORGAN. Yes. It is very difficult to determine with accuracy what contamination we have placed in the neighborhood and what contamination has been placed there from atomic weapons tests because for example, the strontium 90 atom is undistinguishable with reference to the two processes.

So a great deal of radioactivity in the neighborhood areas is undoubtedly due to the weapons tests.

Senator HICKENLOOPER. But a corresponding activity would be noticed at other places, also?

Dr. MORGAN. That is right.

Senator HICKENLOOPER. So it is not a particular environment of Oak Ridge that sees this activity?

Dr. MORGAN. That is right. During this summer a survey was made over the entire United States—by New York Operations Office of AEC—and the levels in the neighborhood of Chattanooga and other nearby cities are essentially the same as those found in other big cities, St. Louis, Dallas, and other places, where they made measurements.

Senator HICKENLOOPER. Now, does the burial of atomic waste solids create a substantial problem?

Dr. MORGAN. Yes; the burial of solids is a problem because I believe we are using of the order of 5 acres a year and it means that we have to provide a permanent burial facility because some of these wastes, at least if they contain plutonium with 24,000-year half life, have to be under surveillance indefinitely.

It also means, of course, that we must make sure of the underground flow of water. In our situation we have the Conasauga shale which retains, which fixes rather permanently, the more dangerous radionuclides. Strontium, practically all of the cesium, and everything with the exception of nitrates and ruthenates seem to be held rather firmly in the soil.

Senator HICKENLOOPER. This material that is buried, you say that requires either periodic or continuous surveillance?

Dr. MORGAN. Yes.

Senator HICKENLOOPER. So that as more material is buried the amount of attention that has to be given to that increases, does it not?

Dr. MORGAN. Indeed.

Senator HICKENLOOPER. Is that apt to become quite a substantial problem as the years go on and we expand atomic energy activities all over the country, I mean the problem of waste disposal?

Dr. MORGAN. The answer to your question is yes, that as time goes on it will take a great deal of earth and a great deal of study on the part of the health physicists and the other engineers in assuring that this material is safely and permanently disposed, both the solid and the liquid waste.

It is for that reason that in our laboratory we are spending quite a large sum in health physics research to explore new and different methods of permanently fixing this material and removing it from the environment. It is our belief that to put it in a tank only postpones the period when you find a more suitable method.

Senator HICKENLOOPER. And, of course, there is an attendant increasing dollar cost in that, in the transportation of it to distant points and explorations for proper places to put it, and so on?

Dr. MORGAN. Yes, indeed. In fact, if you are not careful, the cost of waste disposal can become an appreciable fraction of the cost of your electrical power.

Senator HICKENLOOPER. So that in fact, while we are very anxious for progress in reactor development and its use, the problem of waste disposal will increase with each operation, will it not?

Dr. MORGAN. Right.

Senator HICKENLOOPER. And the same thing that applies here will, I take it, apply every place else in the world? That is wherever they produce these radioactive waste products they have to be disposed of somewhere?

Dr. MORGAN. Yes. This applies particularly at the chemical processing plants. It is thought that in some cases in the future it might be possible to ship the material from reactors to central processing plants.

Senator HICKENLOOPER. I realize that part of the problem revolves around processing plants, but the point I was attempting to explore is that the more atomic operations that we have here and throughout the world, the more waste that has to be disposed of, whether it is disposed of here, there, or some place else, it is an accumulation. There will be a constant accumulation of waste products in direct proportion as the use of atomic fuels and other things grows?

Dr. MORGAN. That is right.

Senator HICKENLOOPER. Is that correct?

Dr. MORGAN. That is correct. So far as we know now there is no way we can decrease effectively the amount of radioactive material produced which, as you say, is produced in proportion to the amount of nuclear power.

Senator HICKENLOOPER. So that if we next year double or triple the number of reactors that are in use, we proportionately increase the problem of disposing of these wastes, that is the burden of disposing of them. I don't know whether it would be three times if we tripled the number of reactors, but it would definitely increase a proportionate amount.

Dr. MORGAN. Yes; and that is why our group, with the support of the Atomic Energy Commission, is exploring many new methods of disposal. We have made quite a study of the permanent fixation by sintering the material placed in the soil. We are beginning to look into

the possibility of placing it in deep wells. The oil well people have used deep wells as a method of disposal of salt water for many years. We are just now actively getting into the study of the possibilities of putting it in salt mines. I am sure that there are other methods, but these are the ones that we particularly are looking into.

Senator HICKENLOOPER. It is all right if it does not leak out of the salt mines.

Dr. MORGAN. You must make sure that it won't; correct.

Senator HICKENLOOPER. Therefore, I take it, we should be also very much interested in waste disposal methods and devices and means of other countries—I say we should be concerned or interested—whether we do anything about it or not is another thing—because they eventually will have large quantities of atomic wastes to be disposed of. I would think that would be a matter of some concern to us, too, the overall worldwide picture.

Dr. MORGAN. Indeed it is of concern to us, but perhaps not as serious a problem as in countries such as France or Germany, where some of the rivers are common to many countries.

But it is encouraging that the International Commission on Radiological Protection, made up of a group of scientists dedicated to the problem of radiation protection is trying to work out these programs and determine the best methods of disposal and the appropriate levels for different situations.

Senator HICKENLOOPER. I think it would naturally follow that it is an increasing problem and will continue to be an increasing problem that will demand solutions that are adequate.

Dr. MORGAN. Yes indeed.

Senator HICKENLOOPER. That is all, Mr. Chairman.

Chairman DURHAM. Dr. Morgan, since waste disposal is a serious problem, you have in operation today what is probably the best disposal plant and the most modern, so far as I know, in the world. I do not know of any operation such as yours. Is that not true?

Dr. MORGAN. Well, I am taking in quite a bit of territory if I say it is the best in the world, or even in the States.

Chairman DURHAM. At Hanford they just bury it and still have it. Since I think this will probably eventually come to this committee, would your recommendation be that we build and try to operate these on the basis of operations, say two or three or four in certain areas, and gather up all the waste and ship it in and not have it strewn about every city or everywhere else as an operating matter?

Dr. MORGAN. I don't think that I would want to recommend that specifically now. I do know, though, that this seems to be a very promising approach to the problem; to find areas peculiarly situated for the disposal of waste and then ship the waste from the large reactors into these sites.

Chairman DURHAM. We would not have every reactor built in the country try to operate a disposal unit.

Dr. MORGAN. That would be extremely unwise.

Chairman DURHAM. Under the circumstances we are building reactors for all over the country already. I think the problem is im-

mediate; not far away. There is no plan so far as I know at the present time except what you are operating, and that is primarily from the operations of the Government that are sent in to you for disposal. Of course, the Savannah River operation receives it, too; is that not correct?

Dr. MORGAN. Yes. Most of the areas are retaining all of their very high level radioactive waste. So far as I know, we have done more exploring in the fixation of this in the natural environment to find more or less permanent methods of getting rid of it. It is for this reason that we are looking into these other methods, because, as the atomic energy program expands you want to pick out those areas that are uniquely adapted to the disposal of waste; so though we have been criticized by some for expanding our efforts on the deep well, the salt mine, and the sintering method, we feel it is important to follow all these methods because you need different methods, any one of which may or may not be uniquely adapted to the particular site.

Chairman DURHAM. I think you are taking the right approach.

Representative HOLIFIELD. I do not think you have any need to doubt that this committee is tremendously interested in this problem which has been brought out by the questioning of Senator Hickenlooper and Mr. Dunham. That is why we are holding the hearings.

We recognize at the present time what we are doing is accumulating atomic cesspools throughout the United States where this highly dangerous material is stored and it is a temporary measure. Anything that can be done toward a permanent solution of this is in my opinion a very urgent matter.

Our so-called Euratom contracts provided for us bringing this material back into the United States for processing in the form of spent fuel rods. So this is bringing back to our shores highly toxic material which is generated in reactors in foreign countries.

In many instances those foreign countries do not have expansive territories. They have, as you pointed out, international drainage conditions which preclude them from handling this material, not to mention the fact that they do not have the separation plants in the quantities that are going to be needed, so as I see it, this is going to become more and more important and more and more dangerous unless there is a permanent solution.

We have some witnesses coming up that are going to describe some of these methods which you have referred to and we hope we will get a good evaluation from them on the possibility of the success of these methods.

Thank you very much. We have run over a bit on this, but it is very important. We appreciate your testimony, Dr. Morgan.

Dr. MORGAN. Thank you.

Representative HOLIFIELD. At this point I would like to place in the record several statements on "Waste Management Operations," at ORNL.

(The statements referred to follow:)

RADIOACTIVE WASTE MANAGEMENT

At

OAK RIDGE NATIONAL LABORATORY

By

F. N. Browder (see p. 514)

I. Introduction

A. General Description

Oak Ridge National Laboratory (ORNL) is a nuclear research and development center operated for the U. S. Atomic Energy Commission by Union Carbide Nuclear Company. The Laboratory employs a total of 4,000 people at two sites, X-10 and Y-12, near Oak Ridge, Tennessee. This report describes the management of wastes from the major radioactive operations at the X-10 site.

The main X-10 area is approximately one mile long by one-half mile wide. Its numerous buildings house laboratories, chemical processing pilot plants, radioisotope production plants, nuclear reactors, and supporting service facilities. Waste collection facilities are located near the center of the main area, and a waste disposal area is situated approximately one mile west of the main area. Figure 1 shows the arrangement of the ORNL X-10 site.

Radioactive wastes are collected and treated according to their physical states (solids, liquids, and gases). To simplify the removal of radioactive components before the wastes are discharged to the environment, the wastes are further classified into the following types for separate collection and treatment:

1. "Highly radioactive" liquid chemical waste: 7,000 gallons per day, 0.001 - 0.02 curie* per gallon, the "hottest" waste at ORNL, which is not highly radioactive compared to high-level wastes at nuclear production plants;

2. Liquid uranium waste (very small volume, moderate radioactivity);

3. Mildly contaminated process waste water - 700,000 gallons per day, 0.1 to 1.0 microcurie* per gallon;

4. Solid waste (variable radioactivity);

5. Reactor cooling air (high radioactivity);

6. Cell and hood ventilation air (low radioactivity);

7. High-radioactivity off-gas.

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*The curie is that quantity of any radioactive nuclide which disintegrates at the rate of 3.7×10^{10} disintegrations per second. This is the unit of radioactive substance equivalent to 1 gram of radium. A millicurie is 0.001 curie; a microcurie is 1×10^{-6} curie.

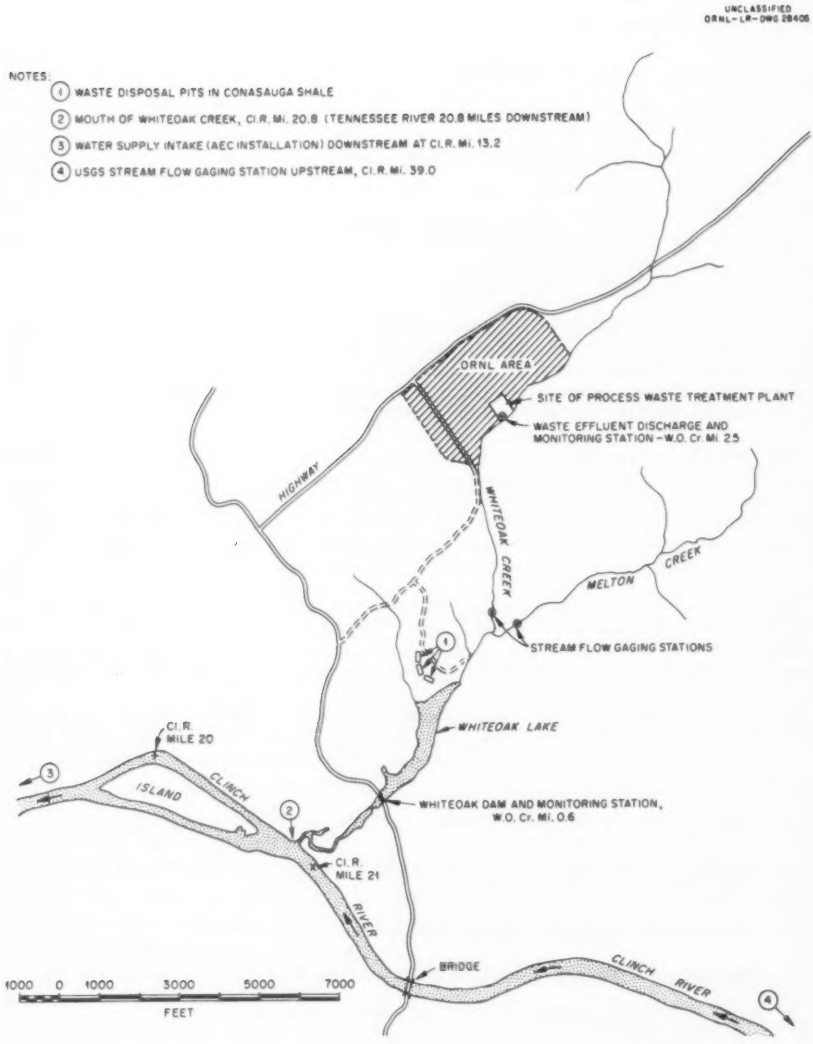
UNCLASSIFIED
ORNL-LR-DWG 28406
NOTES:
1 WASTE DISPOSAL PITS IN CONASAUGA SHALE
2 MOUTH OF WHITEOAK CREEK, CI.R. MI. 20.8 (TENNESSEE RIVER 20.8 MILES DOWNSTREAM)
3 WATER SUPPLY INTAKE (AEC INSTALLATION) DOWNSTREAM AT CI.R. MI. 13.2
4 USGS STREAM FLOW GAGING STATION UPSTREAM, CI.R. MI. 39.0
ORNL AREA
SITE OF PROCESS WASTE TREATMENT PLANT
WASTE EFFLUENT DISCHARGE AND MONITORING STATION - W.O. Cr. Mi. 2.5
HIGHWAY
WHITEOAK CREEK
MELTON CREEK
STREAM FLOW GAGING STATIONS
WHITEOAK LAKE
CI.R. MILE 20
ISLAND
CLINCH
WHITEOAK DAM AND MONITORING STATION, W.O. Cr. Mi. 0.6
CI.R. MILE 21
RIVER
BRIDGE
CLINCH RIVER
1000 0 1000 3000 5000 7000
FEET

Fig. 1 - Location Sketch Map ORNL Area Surface Drainage.

These wastes are described in greater detail below. Figure 2 diagrams the management of these wastes (according to the above number sequence) and shows that after treatment to reduce the levels of radioactivity, liquid wastes are discharged to the Clinch River and gaseous wastes are discharged through tall stacks to the atmosphere. Solid wastes are buried in the ground.

B. Waste Management Philosophy

The goal of radioactive waste management at Oak Ridge National Laboratory is to dispose of the wastes as safely and as economically as possible. At present the following approach to this goal is being taken:

1. To confine the major portion of long-lived fission products from all types of waste in tanks or in the ground, where they will become chemically attached to the soil;

2. To dilute low-level wastes in the atmosphere and the surface water drainage system. The dilution factors available in these media are used to reduce the concentration of long-lived strontium-90 and cesium-137 to the maximum permissible concentrations set by the National Committee on Radiation Protection and the International Commission on Radiological Protection;[1]

3. To monitor the waste streams before and after discharge in order to follow the success of this program.

The disposal of radioactive waste into the ground is considered to be an experiment as well as an expedient solution to a local problem.[2] A number of local conditions permit this ground disposal experiment to be run with apparent safety:

First, the Laboratory is isolated by distance (12 miles to the nearest community) and by patrols that keep the public from potentially hazardous areas. A built-in factor of safety for the public is achieved by the fact that the nearest downstream consumer of Clinch River water is another nuclear plant (the Oak Ridge Gaseous Diffusion Plant--Item 3 in Fig. 1) in the Oak Ridge Control Area, which continuously monitors its water supply for its own protection.

Second, the levels of radioactivity and the volumes of waste from the research and development operations at the Laboratory are lower by at least 1,000 times than those at nuclear production sites such as Hanford and Savannah River.

Third, in the waste disposal area there is a large formation of Conasauga shale containing clays which have a considerable ability to capture and hold radioactive components of waste while allowing the waste water to seep through. The shale provides an inexpensive and effective means of disposal for most of the Laboratory's liquid and solid radioactive wastes. The shale pits are described in Section II, A. Both surface and ground waters from the main X-10 area and the waste disposal area flow to a natural basin drained by White Oak Creek, which empties into the Clinch River within the Oak Ridge Control Area. This restricted drainage provides a means of studying the lateral movement of any radioactive seepage from the waste disposal area.

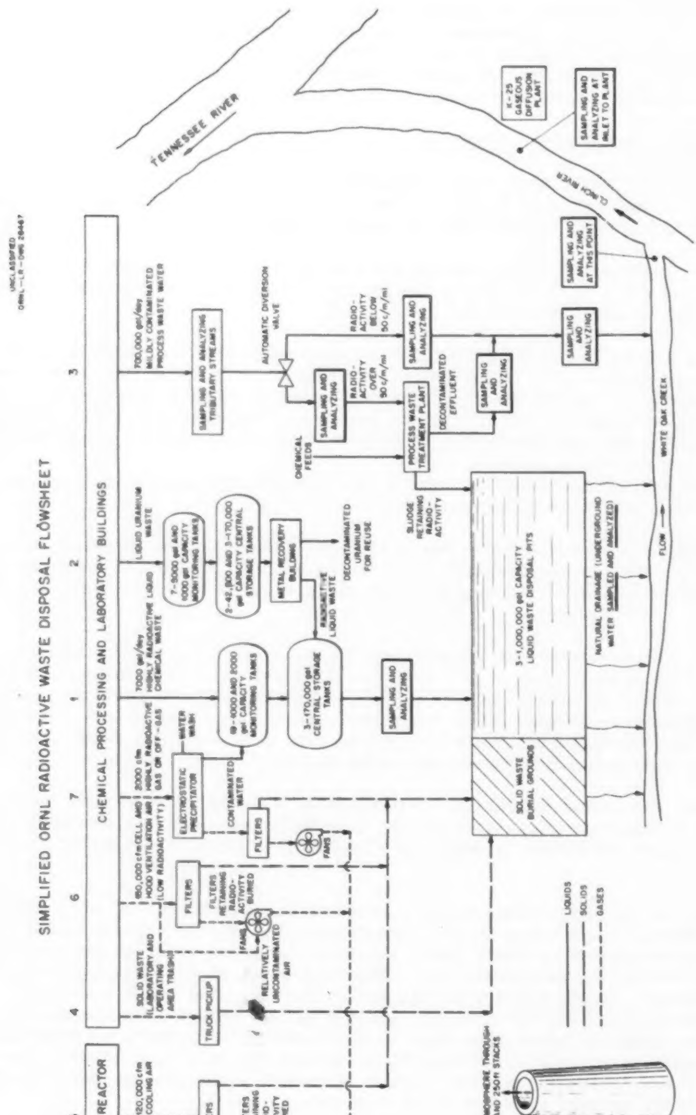

Fig. 2-Simplified ORNL Radioactive Waste Flowsheet

One of the unique factors in dealing with radioactivity is that it cannot be destroyed. No process has yet been developed to speed up or slow down the emanation of particles and rays from radioactive atoms; time is the only agency capable of destroying radioactivity. This fact presents the large quantity producer of radioactive waste with two alternatives for disposing of it:

1. Rendering the radioactive waste harmless by diluting it and disbursing it until the concentration of the radioactive components is too low to be hazardous; or

2. Separating the radioactive components from the waste, concentrating them to a convenient volume, and storing them until time can destroy them by natural decay.

The waste management philosophy at Oak Ridge National Laboratory has been to follow the first alternative for large volumes of low-activity waste and the second alternative for higher-activity wastes.

Responsibility for segregating, collecting, treating, and disposing of all radioactive wastes since the beginning of the Laboratory has been delegated to the Operations Division, which must also measure and record all radioactive contamination as it is discharged to the environment. Responsibility for studying the effectiveness of radioactive waste disposal in the environment by means of stream sampling and surveys and responsibility for reporting the degree of contamination in these streams have been delegated to the Health Physics Division. The separate delegation of these responsibilities provides an independent check on methods of operation and analysis and on concentrations being released to the environment. Research and development efforts on waste treatment and disposal are carried cooperatively by the Chemical Technology and Health Physics Divisions.

C. History of Waste Management at ORNL[3]

The present size of the Laboratory and the scope of its work have been reached by a series of expansions. The means of handling the considerable quantities of waste produced by the processes performed at the Laboratory have had to be changed from time to time to keep pace with the expanding program.

The Laboratory was established in 1943 as a temporary pilot model for the Hanford, Washington, works. The Graphite Reactor, a chemical separations plant (the "Hot Pilot Plant"), and a number of large underground concrete (gunnite) tanks were constructed then. The tanks were intended to store all the highly radioactive liquid chemical waste and the liquid uranium waste accumulated during the life of the Laboratory, which was expected to be one year. However, expansion of the scope of the work in 1943 and indefinite continuation of the Laboratory increased the quantities of waste, necessitating a method of disposal to augment storage in tanks. It was decided to precipitate as much of the radioisotopes as possible in the storage tanks and to decant from the tanks those remaining in solution, dilute them with the Laboratory's large volume of process waste water, and disperse them into White Oak Creek. A portion of the precipitated radioisotopes remain as a sludge in the storage tanks at the present time. A dam was built across White Oak Creek 1.7 miles below the Laboratory (Fig. 1) in the autumn of 1943 to create a controlled area for the discharge of radioactive waste. A settling basin of 1,500,000 gallons capacity was completed in July, 1944, to serve as the waste collection and sampling facility and as a stilling pond to permit radioactive solids to settle from the waste before discharge to the creek.

Additional decontamination of the radioactive supernatant by decay was gained by receiving and holding waste in one of the large storage tanks for as long a time as possible while decanting to the settling basin from another tank containing aged waste. This procedure allowed sufficient time for much short-lived (and hence more intense) radioactivity to decay before the waste was discharged to the creek. The isotopes removed by this procedure were 8-day iodine-131, 28-day cerium-141, 33-hour cerium-143, 41-day ruthenium-103, 12.8-day barium-140, and 40-hour lanthanum-140.

Thus the 7,000 gallons per day of highest activity waste at ORNL was given a precipitation step, about one month holdup for decay, triple settling (in the tanks, the settling basin, and in the lake behind the dam), and about 500,000 to 1 average dilution in the Clinch River. It was calculated at that time that a maximum 5 curies per day of mixed fission products could be discharged safely into the lake, and for several years this criterion was used. An average activity discharge of 284 curies per year, considerably below this level, has been maintained to date (see Table I). This method of disposal by dilution and discharge through a controlled natural drainage basin was continued until June, 1949. This method of disposal had been considered adequate as a temporary measure, but the agreement upon more stringent tolerances necessitated an improvement.

From June, 1949, until June, 1954, the highly active liquid chemical waste was concentrated by evaporation in a pot-type evaporator instead of being decanted and diluted in the lake. During this period the evaporator processed a total of 11,650,000 gallons[4], reducing this volume to 432,000 gallons of radioactive concentrate that were stored in the concrete tanks. The water boiled off from this waste contained an average of only 0.014% of the radioactive contamination entering the evaporator. The effectiveness of the evaporator is demonstrated by the fact that during the period of its operation only 14.5 curies per year came from the evaporator, although an average of 320 beta curies per year was discharged to the creek from the Laboratory. The remaining contamination came from process waste water and from accidental discharges (mainly leaking waste pipes and valves). The evaporator was taken out of service in June, 1954, after the first 1,000,000-gallon experimental ground disposal pit had been in operation for two years. Since that time the pits have received all the highly radioactive liquid chemical waste.

In 1950 the Laboratory was again greatly expanded in size and scope of operation. The waste systems were expanded to handle the increased waste volumes and levels of radioactivity. Monitoring systems for all three liquid waste types were devised to aid the collection and segregation of liquid waste. Underground stainless steel tanks to monitor separately the highly active chemical waste and the liquid uranium waste were installed near each building or area that is a source of either type waste. The tanks permit sampling and measurement of waste volumes and rates of accumulation from each source.

Table I

YEARLY VOLUMES AND RADIOACTIVITY OF DISCHARGES OF LIQUID WASTES

TO WHITE OAK CREEK AT ORNL, 1950-1957[a]

Year	Total Volume, 10^6 gal	Gross Beta Activity, curies			Per Cent of Gross Beta Activity Identified with Specific Radionuclides								
		Settling Basin	Retention Pond	Total	TRE (-Ce)[b]	Ce	Ru	Zr	Cs	I	Sr	Nb	Ba	Co
1950	226.4	172	15	187										
1951	297.6	169	3	172										
1952	268.2	411	87	498										
1953	239.4	289	140	429	53.1	2.6	0.8	2.6	8.5	0.2	27.6	1.0	0.8	---
1954	164.3	237	17	254	34.6	19.1	0.5	1.0	20.3	0.5	23.3	0.4	0.3	---
1955	210.6	213	54	267	30.3	14.7	3.1	0.6	31.6	0.2	18.7	0.6	0.2	---
1956	260.7	253	20	273	24.4	12.3	2.0	0.5	42.1	0.1	15.1	1.0	0.2	2.2
1957	272.3	(combined)		189	25.8	4.4	1.0	10.2	36.5	0.0	18.0	0.7	0.0	1.8

a. Volume of settling basin effluent measured in weir box with liquid level float recorder. For determination of radioactivity, continuous proportional samples of the effluent are composited for daily gross beta measurements and monthly radiochemical separations and analyses.

b. Trivalent rare earths exclusive of cerium.

The monitoring scheme installed for process waste water consists of weirs mounted in manholes in the underground sewer system that collects this waste. These weirs permit measurement of the waste volume from each source and proportional sampling of the waste for determination of the radioactive contamination.

For gaseous wastes a 250-foot stack, filter houses, blowers, underground ducts, and an electrostatic precipitator were added to existing facilities.

In 1951 the first experimental ground disposal pit was built. A larger (1,000,000 gallons capacity) pit was built in 1952, and two more 1,000,000-gallon pits were built in 1955. A pump and a 1-1/2-mile-long pipeline from the collection tank area to the disposal pit area were installed in 1954 to replace a tank truck previously used to transport liquid waste.

Between 1952 and 1957 a metal recovery process reclaimed approximately 130 tons of uranium from liquid uranium waste collected over the years in the original waste storage tanks. The waste from this recovery process was sent to the highly radioactive liquid waste system for evaporation or disposal to pits.

The lake behind the dam on White Oak Creek was drained in 1955 for the following reasons:

1. To perform necessary maintenance work on the dam facilities;
2. To destroy and dispose of the aquatic species in the lake;
3. To avoid attracting and harboring migratory wildfowl;
4. To provide additional safeguards by increasing retention potential;
5. To facilitate and improve control of activity releases;
6. To permit modification of sections of the lake area for research use.

The stream now flows through the lake bed and through a sluice in the dam, which can be closed to impound contaminated water when the need arises. A bypass has been proposed to carry the stream flow around the dam when a need arises to impound water, but this proposal has not been carried out to date. A continuous sampler and a radiation monitor (submerged in a container through which stream water is circulated) have been installed at the dam. The monitor can detect a "slug" of radioactivity and can sound an alarm, but it is considered a stop-gap instrument until a better one can be developed.

Studies of wildlife in the lake were made between 1950 and 1953[5] and in 1956[6,7] to determine some ecological effects of radioactive contamination. Agricultural crops are being grown in the contaminated mud of the lake bed to study uptake of radioactivity by plants[8,9]

In 1957 a waste water treatment plant was completed and put into operation. The function of this plant is to reduce the level of radioactive contamination in the low-activity process waste water discharged to White Oak Creek. An automatic diversion valve is currently (January, 1959) operating to feed the

entire process waste water flow to the treatment plant whenever the level of radioactivity in the waste exceeds a given point. When the activity level is below the set point, the automatic valve diverts the waste around the treatment plant and to the creek through the settling basin.

A multicurie fission product pilot plant was completed in 1957 to recover strontium-90, cesium-137, and other valuable radioisotopes from high-activity liquid wastes. This pilot plant should be most valuable for treating extremely high-level waste from future processes planned for the Laboratory.

II. Current Handling and Disposal Methods for Liquid Wastes

A. Stream No. 1 (Fig. 2): "Highly Radioactive" Liquid Chemical Waste

The term "highly radioactive" is applied to the 7,000-gallon per day liquid chemical waste stream at Oak Ridge National Laboratory only because this stream has the highest level of radioactivity of any waste at the Laboratory. However, compared to highly active wastes (containing 80 to 5200 curies per gallon[10]) at production sites, this ORNL stream is misnamed, because its concentration of radioactive components is normally between 0.001 and 0.02 curie per gallon. In 1957 some experimental operations (Thorex) produced wastes having activity levels as high as 2 curies per gallon when accepted into the waste system, but the total volume of these wastes was less than 1,000 gallons. Because the research and development nature of the Laboratory brings about frequent changes in the processes that produce waste, the waste composition is not consistent. Larger volumes and higher radioactivity levels are expected from future operations. The main radioisotopes are usually cesium-137, ruthenium-106--rhodium-106, strontium-90--yttrium-90, and trivalent rare earth elements. Strontium, cesium,and trivalent rare earths constitute the major fraction of radioactivity on an average disintegration per minute basis. Sodium and nitrate account for about 70% of the nonradioactive solids in the waste.

The handling of highly radioactive liquid chemical waste is illustrated in Figs. 2 and 3, which are simplified flowsheets of the Laboratory's waste systems. The main sources of the liquid chemical waste are chemical processing pilot plants. Radioisotope production facilities and research laboratories produce wastes of smaller volume and lower activity level. The special fission product wastes shown in the upper right corner of Fig. 3 are recovered and consequently do not contribute directly to the waste stream.

Highly radioactive liquid chemical waste is discharged from process vessels in laboratory and pilot plant cells into "hot" drains, which are stainless steel pipes leading to underground stainless steel monitoring tanks. There are now 19 of these 500 to 4000-gal capacity tanks in service (including tank W-1, Fig. 3); their total capacity is approximately 34,000 gal. Each tank is located near its main contributor to permit gravity flow to the tank. The function of these monitoring tanks is to collect the waste and to provide a means of sampling it and of measuring its volume. The tanks provide the Operations Division with a means of checking on waste contributors for rate of waste production and to be sure each type waste gets into its proper system for treatment before disposal. Each tank has connections to the highly radioactive chemical waste system and to the process waste water system to give a choice of treatment depending on composition, radioactivity level and other considerations.

Figure 4 shows the burial details of the monitoring tanks. Each tank is anchored to a concrete saucer that slopes toward a sump. The sump collects ground water seepage and any liquid that leaks from the tank. Liquid in the sump is sampled periodically for radioactivity through a "dry well" to determine whether the tank has leaked. During the past 15 years three tanks were emptied and abandoned after leaks were discovered in this manner. Because most of these leaks were caused by acid corrosion of the tanks, caustic soda to neutralize incoming acid waste is now added to each tank. No leaks have been discovered since this procedure was started.

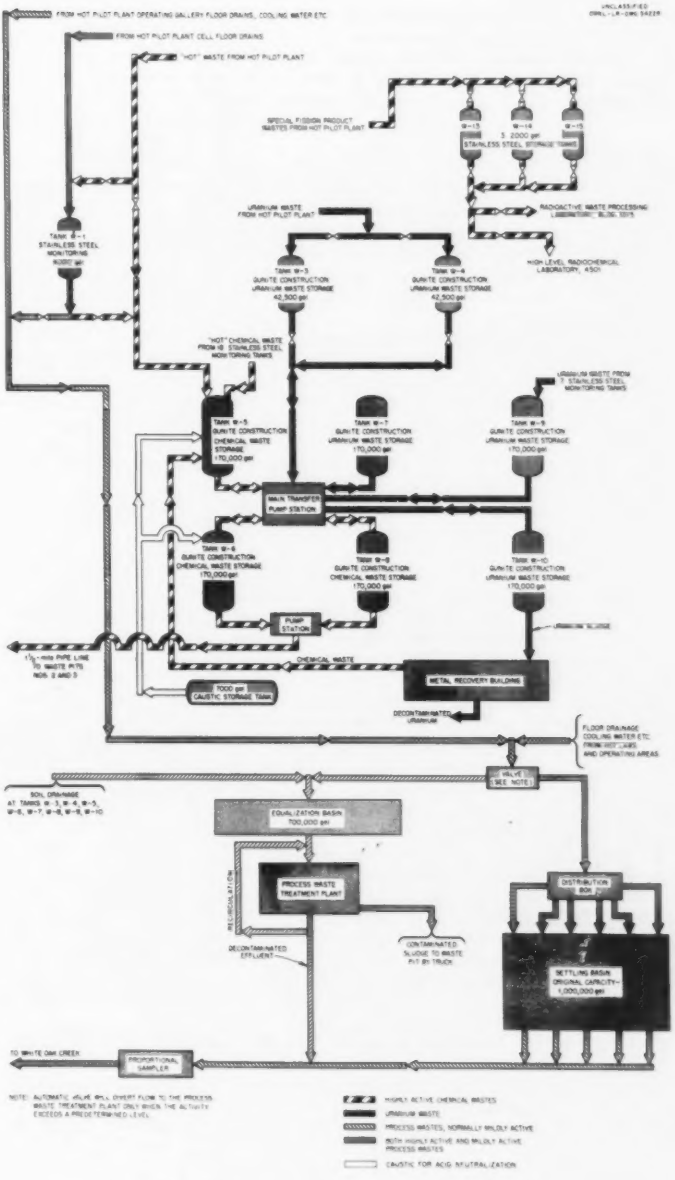
FROM HOT PILOT PLANT OPERATING GALLERY FLOOR DRAINS, COOLING WATER ETC
FROM HOT PILOT PLANT CELL FLOOR DRAINS
"HOT" WASTE FROM HOT PILOT PLANT
SPECIAL FISSION PRODUCT WASTES FROM HOT PILOT PLANT
STAINLESS STEEL STORAGE TANKS
URANIUM WASTE FROM HOT PILOT PLANT
RADIOACTIVE WASTE PROCESSING LABORATORY, BLDG. 3019
HIGH LEVEL RADIOCHEMICAL LABORATORY, 4501
TANK W-1 STAINLESS STEEL MONITORING
TANK W-3 GUNITE CONSTRUCTION URANIUM WASTE STORAGE 42,500 gal
TANK W-4 GUNITE CONSTRUCTION URANIUM WASTE STORAGE 42,500 gal
"HOT" CHEMICAL WASTE FROM 18 STAINLESS STEEL MONITORING TANKS
URANIUM WASTE FROM 7 STAINLESS STEEL MONITORING TANKS
TANK W-5 GUNITE CONSTRUCTION CHEMICAL WASTE STORAGE 170,000 gal
TANK W-7 GUNITE CONSTRUCTION URANIUM WASTE STORAGE 170,000 gal
TANK W-9 GUNITE CONSTRUCTION URANIUM WASTE STORAGE 170,000 gal
MAIN TRANSFER PUMP STATION
TANK W-6 GUNITE CONSTRUCTION CHEMICAL WASTE STORAGE 170,000 gal
TANK W-8 GUNITE CONSTRUCTION CHEMICAL WASTE STORAGE 170,000 gal
TANK W-10 GUNITE CONSTRUCTION URANIUM WASTE STORAGE 170,000 gal
PUMP STATION
1½-mile PIPE LINE TO WASTE PITS NOS. 2 AND 3
CHEMICAL WASTE
7000 gal CAUSTIC STORAGE TANK
METAL RECOVERY BUILDING
DECONTAMINATED URANIUM
FLOOR DRAINAGE COOLING WATER ETC FROM HOT LABS AND OPERATING AREAS
SEE NOTE
SOIL DRAINAGE AT TANKS W-3, W-4, W-5, W-6, W-7, W-8, W-9, W-10
EQUALIZATION BASIN 700,000 gal
PROCESS WASTE TREATMENT PLANT
RECIRCULATION
DISTRIBUTION BOX
DECONTAMINATED EFFLUENT
CONTAMINATED SLUDGE TO WASTE PIT BY TRUCK
SETTLING BASIN ORIGINAL CAPACITY-1,000,000 gal
TO WHITE OAK CREEK
PROPORTIONAL SAMPLER
NOTE: AUTOMATIC VALVE WILL DIVERT FLOW TO THE PROCESS WASTE TREATMENT PLANT ONLY WHEN THE ACTIVITY EXCEEDS A PREDETERMINED LEVEL
HIGHLY ACTIVE CHEMICAL WASTES
URANIUM WASTE
PROCESS WASTES, NORMALLY MILDLY ACTIVE
BOTH HIGHLY ACTIVE AND MILDLY ACTIVE PROCESS WASTES
CAUSTIC FOR ACID NEUTRALIZATION

Fig. 3 - ORNL Simplified Liquid Waste Disposal Flowsheet.

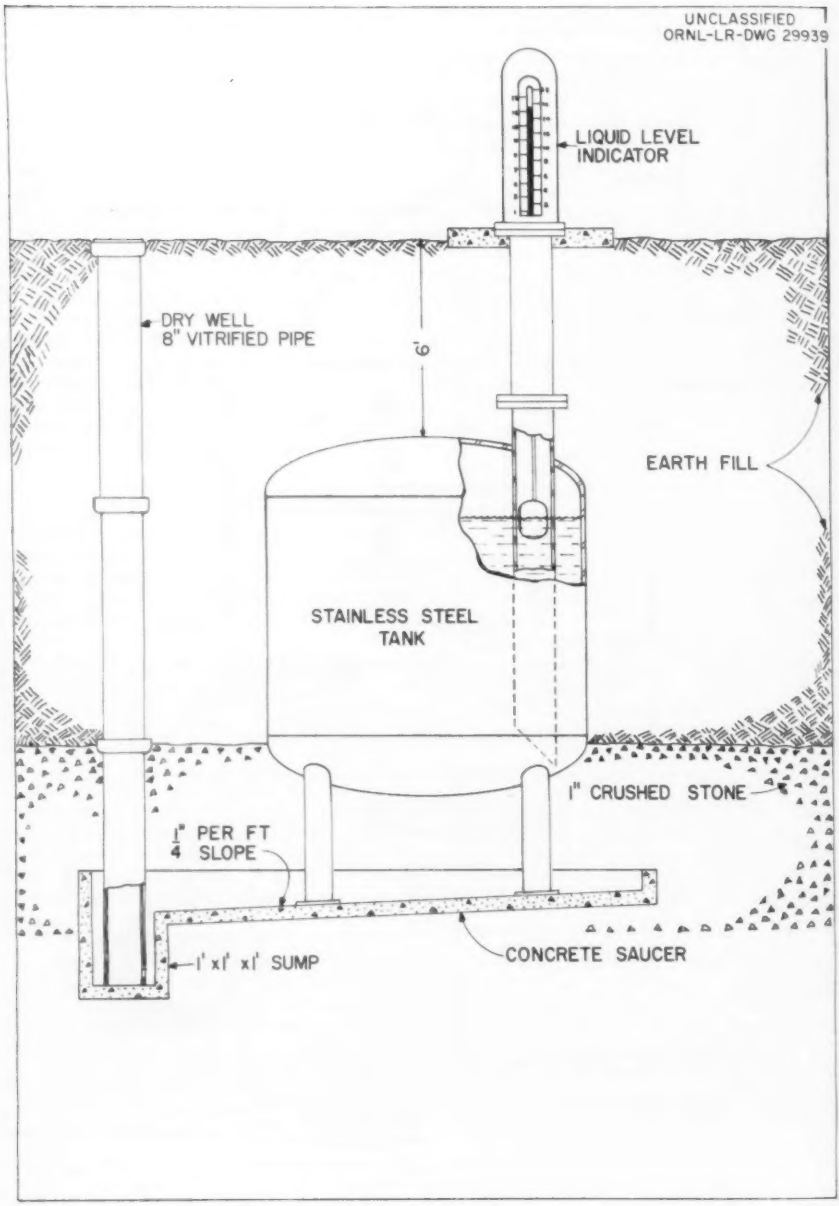

Fig. 4 - Dry Well and Waste Collection Tank Installation.

Each tank is equipped with a float-type volume gauge, which shows on a board above ground the quantity of waste in the tank. Although the rate of waste accumulation in all 19 monitoring tanks is continuously recorded at a central station by a telemetering system, a waste system operator visits each tank every four hours to check the waste volumes lest a tank overflow because of a failure in the telemetering system.

When a monitoring tank becomes filled, the operator switches on a self-priming, packless pump to transfer the waste to one of the three 170,000-gallon concrete storage tanks in the storage farm centrally located in the main X-10 operating area. These tanks rest on concrete saucers that drain to a system of dry wells for leak detection similar to those of the monitoring tanks. The central storage tanks also have volume gauges, samplers and an underground system of transfer lines that permit transfer of the wastes from one tank to another.

The function of the central storage tanks is to provide temporary storage for the highly radioactive liquid chemical waste while the short-lived radioisotopes decay (see section I, C). These tanks provide adequate surge capacity to accommodate normal accumulation from Laboratory operations even during occasional shutdown and repair of pumping equipment. Wastes are periodically pumped from the central storage tanks approximately a mile and a half through a 2-in. dia underground steel pipe to the disposal pits.

The waste pits are three 1,000,000-gal open cavities bulldozed in the earth in a location chosen for remoteness from the Laboratory, the type soil, and the fact that underground drainage is toward White Oak Creek.[2,11a] The soil, Conasauga shale, has the property of removing and retaining most of the radioactive components while the waste water and certain nonradioactive chemicals seep through it toward White Oak Creek. Waste enters pit No. 3, which overflows through a valved pipe to pit No. 2, which overflows similarly to pit No. 4. Figure 5 is an aerial photograph of the waste disposal area showing the pits in the foreground, the creek, and in the upper left corner a part of the main X-10 area. The pits are each 15 ft deep with sides sloping at an angle of 30°. Their top dimensions are 210 ft by 100 ft. The pits are covered with wire screen to prevent access to wild life. The waste discharged into the pits is sampled and analyzed for radioisotopes and stable chemical ions, and the movement of these materials in the soil and in the seepage into the creek is monitored by the Health Physics Division. The only radioisotopes detected in the seepage to date are ruthenium-106, cobalt-60, and antimony-125. By the end of 1956 the seepage of ruthenium-106 was estimated to average 23 curies per year. In 1957 samples indicated 200 curies of ruthenium-106 seeped from the pits, but only 60 curies of ruthenium-106 were detected passing White Oak Dam. In 1957 a total of 41,900 curies was discharged to the pits, an increase of 20% over 1956; in 1958 52,800 curies were discharged, an increase of 25% over 1957. These increases were due mainly to higher radiation levels in processing at the Hot Pilot Plant. By the end of 1958 the total of all wastes discharged to the pits since the start of this practice was 11,703,000 gallons containing 167,000 curies (at time of discharge).

B. Stream No. 2 (Fig. 2); Liquid Uranium Waste

The quantity of liquid uranium waste is much less than it was earlier in the history of the Laboratory, as it is now produced at a maximum rate of only 100 gallons per week. In the sense that it is actually a solution of reusable uranium

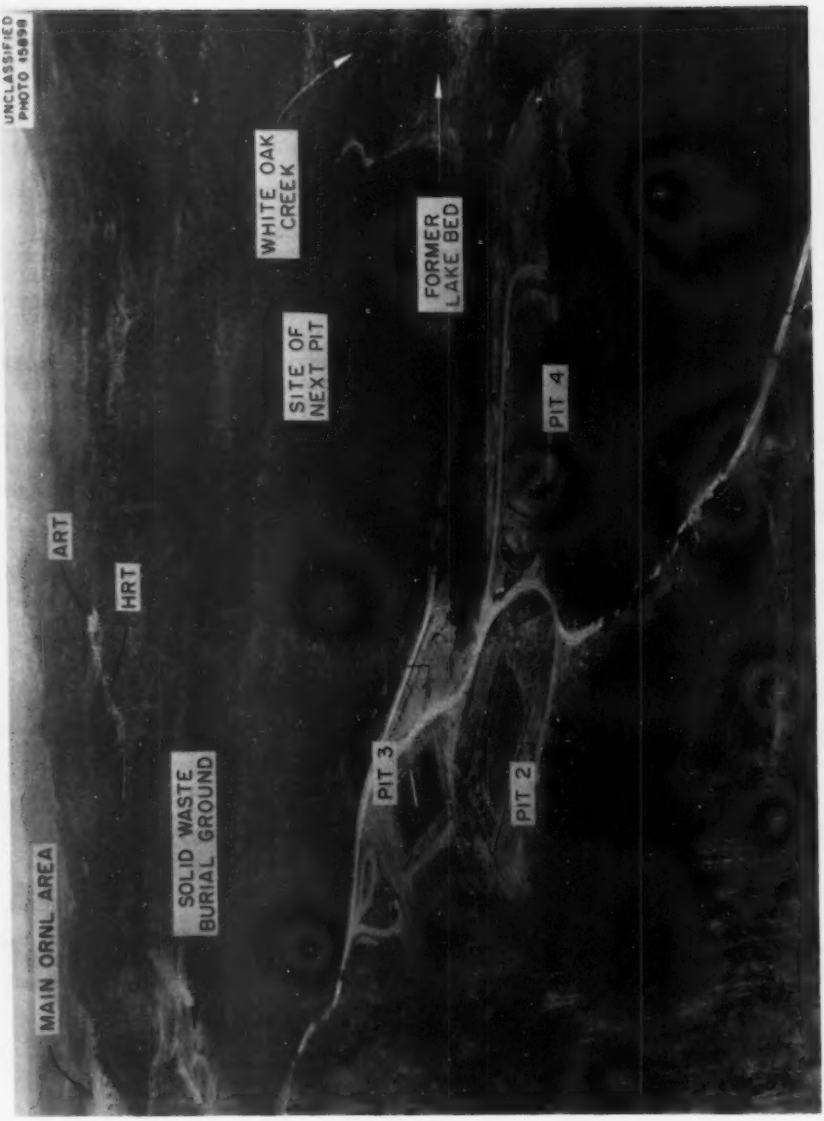

Fig. 5-ORNL Waste Disposal Area.

contaminated with fission products, this is not a true waste. It is collected and stored temporarily prior to treatment for recovery of the uranium. Between 1952 and 1957 approximately 130 tons of uranium were recovered. The separated fission products are discharged to the highly radioactive chemical waste system. Figures 2 and 3 illustrate the handling of liquid uranium waste.

The liquid uranium waste from the various contributors is collected separately from other liquid waste by means of a system of seven monitoring tanks (total capacity 7,500 gallons), similar in all respects to those described above for the highly radioactive liquid chemical waste. Each buried tank serves a source of liquid uranium waste, which flows by gravity through special stainless steel drains from the source facilities to the tanks. The waste volumes in the tanks are recorded at the central station by the previously described telemetering system, and a waste system operator checks the tank gauges every four hours to prevent overflows. The waste from full tanks is pumped to the central tank farm, where three 170,000-gallon and two 42,500-gallon underground concrete tanks store it until enough has accumulated to warrant running the uranium recovery facility.

C. Stream No. 3 (Fig. 2): Process Waste Water

The sources of process waste water are equipment cooling systems, floor drains, decontamination pad drains, storage canals, laboratory sinks, and discharges from low activity operations. It is the least radioactive of all Laboratory liquid wastes except sewage and storm water, yet it is the most difficult to manage because of its combination of radioactivity and large volume. The 700,000 gallons per day of this waste make storage impractical and necessitate disposal on a current basis. The waste is collected, sampled and discharged to the creek continuously. Figures 2 and 3 illustrate the relation of the process waste water system to the other liquid waste systems.

The process waste system sometimes serves as an emergency "warm" system. Much of the radioactive contamination put through this system is a result of equipment failure, human error, or accidents that cause a misdirection of contamination from the highly active chemical waste system. Whenever unusually high levels of radioactivity occur, an effort is made to divert the radioactive portion to the high activity level system as soon as possible. A network of 6-in. to 30-in.-diameter vitrified clay pipes collects and conveys the process waste water by gravity flow to a central monitoring point near the inlet to the 1,500,000-gallon settling basin, where the volume of flow is measured and sampled continuously. The samples are collected every four hours and analyzed for gross beta activity. The process waste water collection system is divided into several sections, each of which is served by a strategically located monitoring station. Each station is a concrete manhole in which are mounted a V-notch weir, a water level recorder for determining volume of flow and a finger-type pump for collecting samples whose volumes are proportional to the flow. When the radioactivity level at the central monitoring station rises to about 150 c/m/ml* or greater, the source

Fig. 5-ORNL Waste Disposal Area.

*c/m/ml = counts per minute per milliliter. All results are referred to second-shelf counting using an end-window Geiger counter.

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responsible for the increase can be located by referring to the monitoring samples, and corrective action can be taken to minimize discharges of radioisotopes through this low-level system. With a new treatment plant available, process waste water can be temporarily held and treated to reduce the level of activity. The treatment plant is described in more detail below.

The yearly volumes and radioactivity of the process waste water during the years 1950 to 1957 are summarized in Table I. The total volume of flow has ranged from about 165 to 298 million gallons per year for an average of 700,000 gallons per day. The level of gross beta activity normally ranges from less than fifty to several hundred c/m/ml with occasional transient levels above 1000 c/m/ml, depending upon operating conditions in the Laboratory. The fourth column in Table I, headed "Retention Pond," refers to a small volume of drainage from the monitoring pads underneath and from the soil around the underground waste storage tanks in the central tank farm. This pond previously served as a monitoring point for the detection and measurement of any leakage in the highly radioactive liquid chemical and liquid uranium waste piping and storage systems. The retention pond stream has been intercepted and is now pumped to the equalization basin of the new process waste treatment plant. The total gross beta activity of discharges to White Oak Creek from the process waste water system and the retention pond has ranged from 172 to 498 curies per year. Despite a hundred-fold increase in levels of radioactivity in the operations at ORNL since 1951, Table I shows that the activity levels in discharges to the creek have not increased proportionally. The 1957 low of 189 curies was achieved by a reduction in the frequency of accidental releases, constant policing of process waste streams, and operation of the process waste water treatment plant. Less than 100 curies were discharged in 1958. The waste discharged from the collection system is normally clear with very little suspended matter; but it varies widely in acidity or alkalinity, the pH ranging from about 2.0 to 11.5.

The flow of the effluent from the settling basin is measured at a weir box near the point of discharge to the creek, and composite samples are collected by a Trebler proportional sampler. Radiochemical analyses of these samples for five years (1953-1957) are summarized in Table I. The chemical, radioactive, and physical properties of the waste discharged from the settling basin may be influenced by sedimentation in the basin or by heavy growths of algae, both of which are capable of concentrating radioactivity.

Process Waste Water Treatment Plant[12, 13, 14]

Increases in the chemical processing operations and in their radioactivity levels at ORNL during several recent years increased the volume and activity level of process waste water discharged to the creek, as shown in Table I. On several occasions surveys revealed that the level of radioactivity in the Clinch River for short periods exceeded the recommended average concentration limit for unidentified radionuclides.[1] These incidents emphasized the need for a treatment plant to reduce the level of radioactivity in the process waste water. The low levels of radioactive discharge in 1957 and 1958 demonstrate that the treatment plant operation plus monitoring efforts within the Laboratory have been successful.

The radioactive contamination discharged to the creek has always passed through the process waste water system as a very dilute solution chemically similar to "hard" water. The over-all results of extensive laboratory and pilot

plant studies indicated that the process waste water treatment plant should be designed to use a horizontal flow, lime-soda water softening process with provision for alternative use of phosphate coagulation to effect strontium removal and the addition of clay to increase the removal of cesium.

The plant, completed in August of 1957, is located near the central monitoring and diversion station at the outlet of the process waste water collection system (Fig. 1). It has a design capacity of 500,000 gallons per day, and provision has been made for future expansion to double this capacity. In addition to the routine treatment of waste, the plant is designed to give special treatment to waste containing abnormally high concentrations of radioisotopes caused by accidental or emergency releases.

Figure 6 is a diagram of the facilities for treatment and disposal of process waste water. When the inlet valving is operated manually, the plant receives all process waste water flowing between 8:00 a.m. and 8:00 p.m., the period when most of the contaminated discharges occur. This method results in (1) the treatment of large volumes which might not need decontamination and (2) the possible bypassing of active wastes at night. Therefore, an automatic diversion valve was provided which, in response to submerged Geiger-Mueller tubes, permits low level wastes to bypass the plant but diverts into the equalization basin all wastes which contain a greater concentration of radioactivity than a predetermined amount.[15] The 700,000-gal capacity equalization basin serves to minimize fluctuations in the composition of the plant influent and to supply waste to the treatment plant as needed.

Figure 7 is a cut-away view of the process waste water treatment plant. Uniform flow rate through the plant is maintained by two 350-gpm centrifugal pumps drawing from the equalization basin. Two gravimetric feeders apply slurries of lime and soda ash to the flash mixer which has a detention time of 1.5 minutes. (A third feeder is available for clay, trisodium phosphate, etc., as needed.) The three coagulation basins in series, providing 30 minutes of slow mixing, are followed by 2 hours settling in a 12 x 70 x 8 ft deep basin. The effluent is discharged to the creek. It can be recycled through the plant for additional treatment, although this has not yet been attempted to date.

The sludge that precipitates in the settling chamber contains the radioisotopes removed from the waste. A sludge scraper operating continuously moves it along the concrete bottom of the settling chamber into hoppers at the deep end of the chamber. Valves permit the sludge to drain from the hoppers to a 16,000-gallon concrete tank for short-time storage before disposal. Two plunger-type pumps lift the sludge to a partly shielded tank truck, which transports it to the disposal pits for highly radioactive chemical waste.[2] A part of the sludge can be recycled through the treatment plant when this is desirable.

The mechanical equipment is designed to allow maintenance without draining the chambers, thus utilizing the waste water as a shield to protect personnel from exposure to radiation from the radioactive sediments in the chambers.

During the first 11 months of operation 53 to 87% of the gross beta activity was removed by a single pass of the waste through the treatment process. The removal of strontium-90, considered to be the most hazardous component, ranged

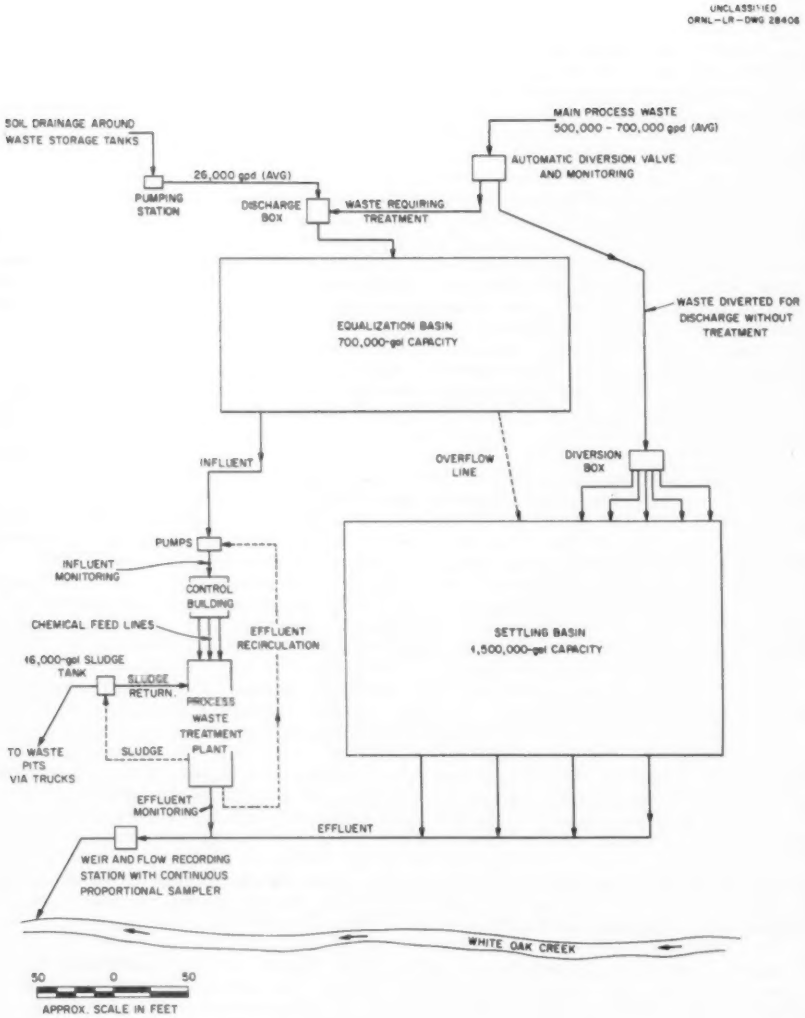

Fig. 6- Flow Diagram of Process Waste System.

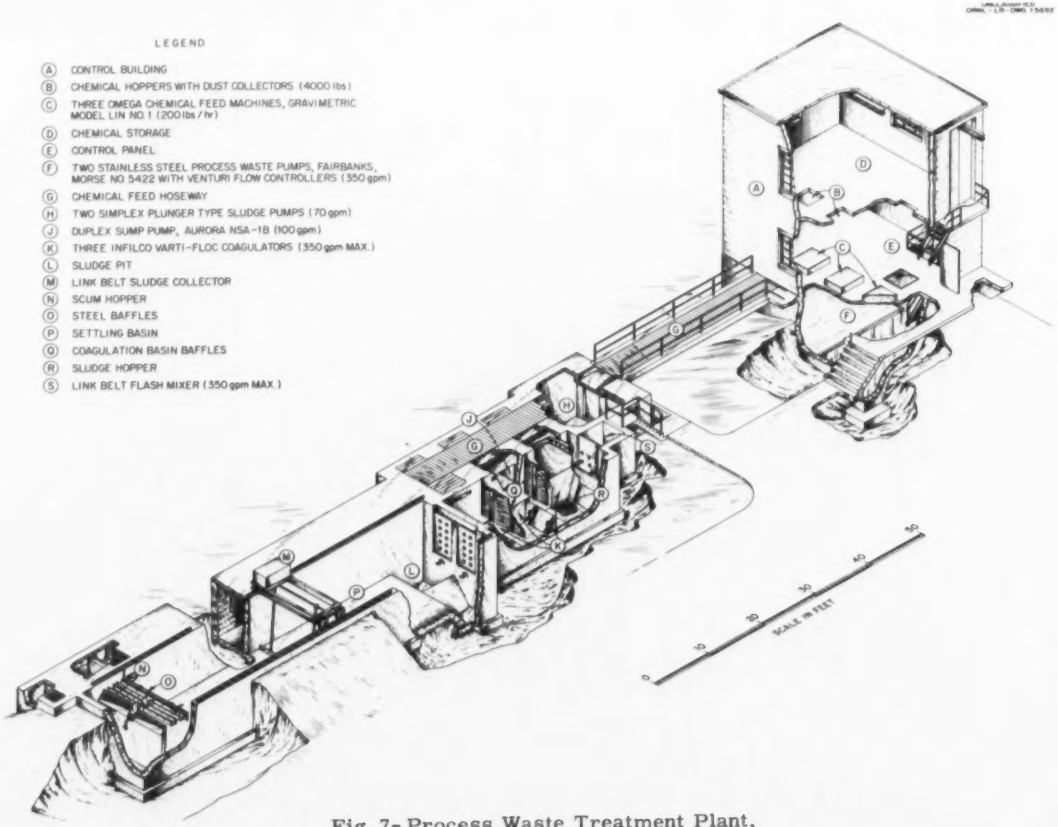
LEGEND
(A) CONTROL BUILDING
(B) CHEMICAL HOPPERS WITH DUST COLLECTORS (4000 lbs)
(C) THREE OMEGA CHEMICAL FEED MACHINES, GRAVIMETRIC MODEL LIN NO. 1 (200 lbs / hr)
(D) CHEMICAL STORAGE
(E) CONTROL PANEL
(F) TWO STAINLESS STEEL PROCESS WASTE PUMPS, FAIRBANKS, MORSE NO. 5422 WITH VENTURI FLOW CONTROLLERS (350 gpm)
(G) CHEMICAL FEED HOSEWAY
(H) TWO SIMPLEX PLUNGER TYPE SLUDGE PUMPS (70 gpm)
(J) DUPLEX SUMP PUMP, AURORA NSA-1B (100 gpm)
(K) THREE INFILCO VARTI-FLOC COAGULATORS (350 gpm MAX.)
(L) SLUDGE PIT
(M) LINK BELT SLUDGE COLLECTOR
(N) SCUM HOPPER
(O) STEEL BAFFLES
(P) SETTLING BASIN
(Q) COAGULATION BASIN BAFFLES
(R) SLUDGE HOPPER
(S) LINK BELT FLASH MIXER (350 gpm MAX.)
0
10
20
30
40
50
SCALE IN FEET

Fig. 7- Process Waste Treatment Plant.

from 66 to 87% for monthly periods and has been as high as 94% during one week. The trivalent rare earth elements were removed as efficiently as the strontium; about 17 curies of strontium and 15 curies of total rare earths were removed from the process waste during this period. Cesium removal has been low because clay has not been added to the process on a continuous basis, but during one week 30 ppm of a local clay increased cesium removal from 16% to 55%. (A commercial source of a similar clay suitable for continuous use has been located.) The operating experience to date indicates that this facility will fulfill the primary purpose of increasing the safety factor in control of radioactive waste released to the Clinch River.

D. Wastes Discharged Directly to Creeks

There are several comparatively minor sources of radioactive waste not included in the main ORNL liquid waste control systems listed in Fig. 2 because of remote location or other considerations. Normally these sources produce low activity waste most of which is monitored before release from the source facility to the nearest creek. These sources are:

The Low Intensity Test Reactor (LITR). Two retention ponds catch the primary coolant (water) when the reactor is drained or when it leaks coolant. Cooling water from irradiation experiments in the reactor also is discharged to these ponds. Most of the radioactivity is caused by sodium-24, which decays rapidly. The ponds are periodically drained to a branch of White Oak Creek only after the water radioactivity has decayed to the extent that a scintillation counter reads lower than 100 gross beta and gamma counts per minute per milliliter. The LITR can discharge its higher activity waste to a monitoring tank in the highly radioactive chemical waste system previously described.

The Oak Ridge Research Reactor (ORR). The LITR retention ponds also catch reactor water drainage from the "warm" sump in the ORR building. Higher level wastes can be handled through the same monitoring tank that serves the LITR.

The Homogeneous Reactor Test (HRT). The HRT facilities are located in Melton Valley across a high ridge from the main X-10 area (Fig. 5). A retention pond, excavated in Conasauga shale near the reactor building receives reactor storage-canal water and cell shield water when these chambers are drained. The pond overflows to Melton Branch, a tributary of White Oak Creek (Fig. 1). The capacity of the pond equals the entire volume of all the HRT building cells, which are filled with water to shield personnel when repairs to equipment in the cells are required. The level of radioactivity in the water should normally be quite low. In the event the activity level is high enough to make discharge to the pond undesirable, the water can be drained to a 12,000-gallon stainless steel tank buried near the pond. After concentration by an evaporator this waste can be hauled in shielded carriers to the highly radioactive waste disposal pits previously described. The evaporator effluent can be recycled or discharged to the pond.

The Aircraft Reactor Test (ART). The ART facilities are located away from the main ORNL area and near the HRT site in Melton Valley. Although the facilities are not now in use, they can be used for radioactive operations. Drains in the reactor building lead to a sump from which low level waste can be pumped to a retention pond. This pond is excavated in the shale near Melton Branch about 300 yards from the reactor building. No provision has been made to drain or to overflow the pond.

The Tower Shield Facility (TSF). The TSF, which contains two water-cooled reactors, is located near the Clinch River two miles from the main X-10 area. An open pool 20 ft square by 25 ft deep and an adjoining equipment storage basin 4 x 8 x 12 ft deep serve these reactors. The water circulates through one of the reactors and returns to the pool, which periodically overflows downhill away from the facility. A one-gallon sample is taken from the pool once a week for radiation analysis, and the mud on the hillside is sampled when overflows occur. Sodium-24 and irradiated water impurities comprise almost all the radioactive contamination in the water. Four 20,000-gallon tanks are available at the TSF site for contaminated water holdup in the event fission products get into the water from a ruptured fuel element.

The other reactor has a closed-circuit cooling system equipped with a water demineralizer. Most of the radioactivity will be caught and will decay in the demineralizer, which will be drained to the pool periodically.

The ORNL Laundry. The contaminated-clothing laundry discharges its waste water directly to White Oak Creek. Normally this waste is not monitored before discharge because the clothing is monitored to avoid washing of highly contaminated material and because contamination is diluted further by the washing process.

The ORNL Rolling Mill. The rolling mill, where uranium fuel elements are fabricated, has floor drains that discharge directly to a branch of White Oak Creek. Rolling mill waste is not normally monitored before reaching the creek.

Fallout from ORNL Gas Disposal Stacks. Radioactive particles from gas disposal operations can fall throughout the ORNL drainage area and thus get into the streams. Continuous monitors measure the fallout, and rain gauges at the air monitoring stations gather samples for radioactive analysis. The air monitoring program is described in section V, A.

Although the very dilute wastes discharged directly to the creeks, as described in this section, are not included in any of the main ORNL liquid waste control systems, monitoring of the creeks downstream affords a reasonable measure of control. The stream monitoring program is described in section V, B.

III. Current Handling and Disposal Methods for Gaseous Wastes

Air contaminated with radioisotopes constitutes the greatest volume of radioactive waste but not necessarily the greatest hazard created by the operation of nuclear reactors and associated chemical processes and laboratories. Basically, there are three general types of air-borne contaminants. The nature of the hazard involved and the technology required for its elimination differ for each type:

1. Radioactive noble gases, xenon, krypton, etc., which are chemically inert, do not constitute a serious hazard at ORNL and are controlled by releasing the contaminated air from tall stacks for dilution by the atmosphere. (400 curies per day of argon-41 are discharged).

2. Chemically reactive gases, such as I_2, are controlled by adsorption equipment (usually caustic scrubbers) located in the process areas.

3. Radioactive solid and liquid particles suspended in air are controlled effectively by filters and electrostatic precipitators.

All air that has a possibility of becoming contaminated is exhausted to the atmosphere from three tall stacks for dispersal. Standards for safe discharge are set and released waste gases are monitored by the ORNL Health Physics Division. The air passing through the stacks is continuously sampled and analyzed, and the fallout is continuously checked at 17 locations within seven miles of the Laboratory. One station (Berea, Kentucky) is about 120 miles away. The volumes of air and their flow patterns through the buildings are controlled to minimize the amount that has a possibility of becoming contaminated. Contamination of large quantities of air surrounding process equipment is avoided by designing the equipment to be airtight and by operating all process under a negative pressure. The air-borne radioactivity released during processing is thus caught in a relatively small amount of air, called "vessel off-gas", which is treated before release. The air surrounding the vessel in concrete cells, designated "cell ventilation gas," thus has a small possibility of becoming contaminated and is handled on a different basis from the vessel off-gas. These methods of handling the large quantities of air passing through the Laboratory facilities sharply reduces the amount of air requiring treatment before discharge.

The types of air handled at ORNL and whether or not each is treated before discharge can be briefly summarized as follows:

Office Ventilation: Six changes of air per hour, without treatment;

Laboratory Ventilation: Minimum of ten changes of air per hour, without treatment;

Laboratory Hood Ventilation: Minimum of 50 feet per minute face velocity for adequately designed hoods, with provisions for the installation of a filter in the discharge duct when necessary. In addition, each hood is equipped with two vacuum systems, the first system with 10 inches of water vacuum to draw gases from vessels containing high levels of radioactivity and the second system with 20 inches of mercury vacuum to be used for solution transfers and other

high vacuum applications. The air from both these vacuum systems is collected and treated as vessel off-gas before being discharged to the atmosphere.

Cell Ventilation: Minimum possible air flow; each cell is held at a reduced pressure (up to 1 in. of water vacuum) so that air leakage will be inward. The air drawn from cells where there is a high risk of its becoming contaminated (usually cells containing pilot plant or experimental equipment) is filtered. All cell ventilation air is discharged from the tall stacks for dispersal to the atmosphere.

Vessel Off-gas: Minimum possible air flow in a system with a vacuum of 40 inches of water for dissolver and process vessel off-gas lines. The air from this vacuum system is treated before discharge.

Reactor Cooling Air: All air used in cooling a reactor is filtered.

Air Cleaning Facilities

A central gas disposal facility has been established at Oak Ridge National Laboratory to clean the radioactive contaminated air from the chemical processing areas and to dispose of it to the atmosphere.

Vessel Off-gas Facilities; Stream No. 7 (Fig. 2): The air from the vessel off-gas system is moved by high speed stainless steel exhausters through stainless steel ducts and collected at the central gas disposal facility. Reactive gases, such as iodine, are removed by caustic scrubbers in the processing buildings before the air enters the ducts. The air is cleaned by passing it through a Cottrell electrostatic precipitator followed by a filter identical in performance with those described below for the Graphite Reactor. This system has a capacity of 2000 cubic feet of air per minute and collects gas from all radioisotope production vessels, off-gas from hoods, and ventilation air from dry boxes. The cleaned air is discharged to the atmosphere through a 250 ft stack. A proposed cleaning system to handle an expected increase of 4000 cubic feet per minute of contaminated air is in a preliminary design stage. This new system will handle highly radioactive gases from the Oak Ridge Research Reactor (ORR) experiments as well as from future expansions of the chemical processing Hot Pilot Plant. A caustic scrubber plus high efficiency filters will probably be the cleaning equipment used.

The Cottrell precipitator is of the exposed-tube type, containing twenty-three 8-in.-dia tubes 12 ft in length, fabricated of No. 14 gauge stainless steel. Those parts of the precipitator that come in contact with corrosive gases to be cleaned are constructed of type 347 stainless steel. The discharge electrode system is stainless steel wire longitudinally centered through each collecting electrode, suspended from porcelain insulators and held taut by porcelain weights at the bottom.

The precipitator is equipped with a continuous water-flush system. Nozzles are provided at the top of each tube in such a manner that a continuous film of water is maintained on the inner surface of the tube. In addition, a water-spray system is located at the extreme top to facilitate washdowns. The water is collected in a catch tank and recirculated to the precipitator. When the activity level or the acid content (NO and NO_2 are present in the off-gas) becomes high,

the water is discharged to a monitoring tank in the highly radioactive liquid chemical waste system.

The precipitator is designed to operate at a maximum temperature of 200°F and a negative pressure of 60 inches water at a flow of 2000 cubic feet per minute. The water flush system operates at 3 to 6 gallons per minute per tube.

The electrical design voltage is 75,000 volts, 25 kilovolt-amperes and operates on 440-volts 3-phase 60-cycle current to the power pack. In actual practice the precipitator operates on 52,000 volts and a current of 130 milliamperes.

Filtered Cell Ventilation Facilities; Stream No. 6 (Fig. 2): Ventilation air from radiochemical processing areas is collected at the central gas disposal facility through steel and concrete ducts. The air is moved through the ducts by high capacity, low vacuum exhausters. The cell ventilation cleaning facility consists of a bank of Fiberglas filters backed by high efficiency paper filters and is essentially a miniature of the Graphite Reactor filter house described below. An additional filter bank, similar in filtering detail, has just been constructed in another area to filter cell air from the Hot Pilot Plant. All cell ventilation air is dispersed by a 200-ft-stack.

Unfiltered Cell and Hood Ventilation Facilities; Stream No. 6 (Fig. 2): A system of steel and concrete ducts collects ventilation air from laboratory hoods and experimental cells in which the possibility of air contamination is slight. This air is dispersed to the atmosphere via a 200-foot stack without preliminary cleaning. High capacity, low vacuum exhausters move the air through the ducts.

Figure 8 is a photograph of the central gas disposal area. To the right behind the concrete barricade is the filter bank for the cell ventilation system. In the left foreground are the main exhaust fans. Figure 9 is a photograph of the same area showing the top of the Cottrell precipitator and the stainless steel discharge line to the stack. Figure 10 is a photograph of the precipitator during construction.

In addition to the main central facility, several small air-cleaning filters are located at isolated experimental radiochemical installations. Except for one unit in which a graded filter-fiber unit developed at Hanford is used, these are minature replicas of the Graphite Reactor filter installation.

ORNL Graphite Reactor Cooling Air; Stream No. 5 (Fig. 2): For cleaning the cooling air from the nuclear reactor, consideration was given to several methods including cyclone separators and electrostatic precipitators. Filtration was selected as the method to be used after extensive literature research and consultation with specialists in air cleaning problems. The short-lived radioactive gases are not removed from the air before discharge to the atmosphere. The filter house is a large reinforced concrete structure composed of four identical cells, each containing a roughing filter and a polishing filter. The air from the reactor enters the top of the filter house, passes downward through the roughing filter, then horizontally through the polishing filter into the exit air duct. Two fans, operating in parallel at a negative pressure of 50 in. water gauge deliver 60,000 cubic feet per minute each of air to a 200-ft stack. The expected

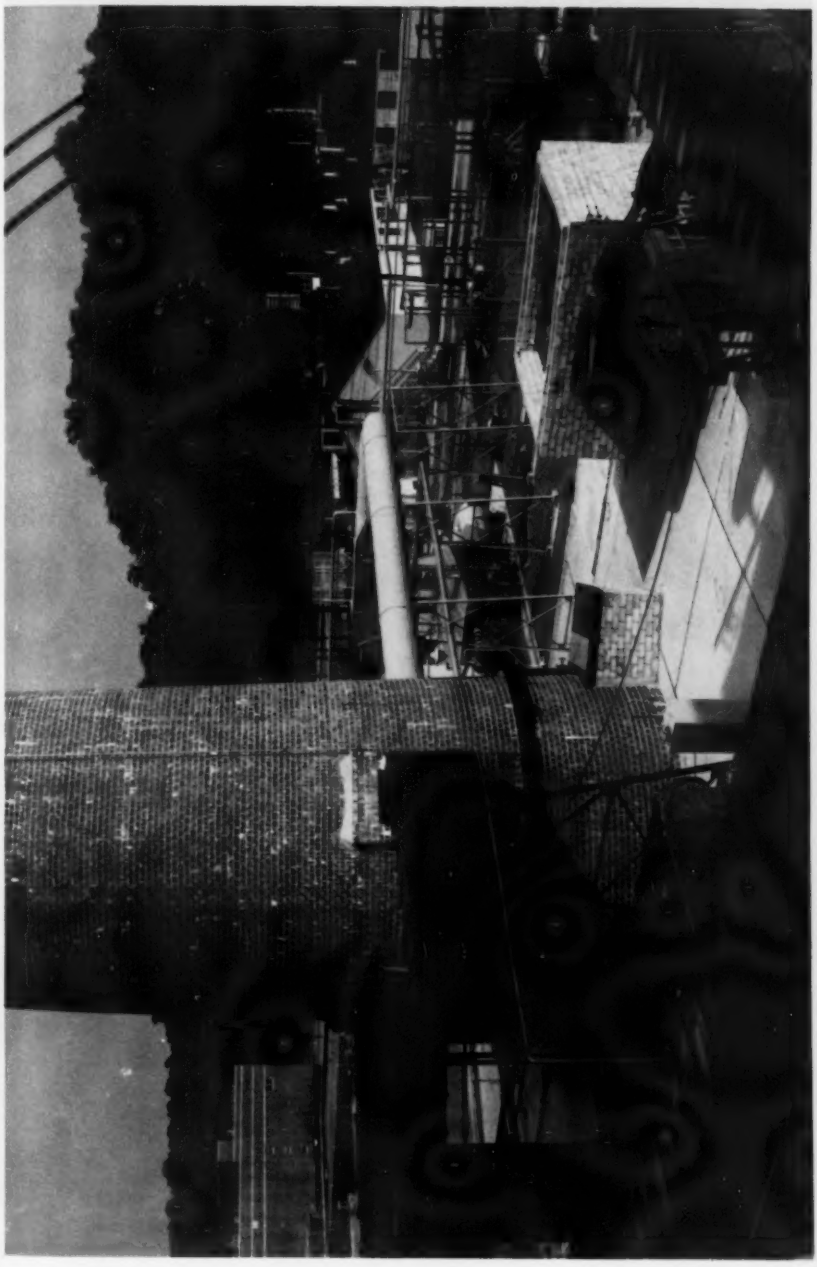

Fig. 8 - Central Gas Disposal Area.

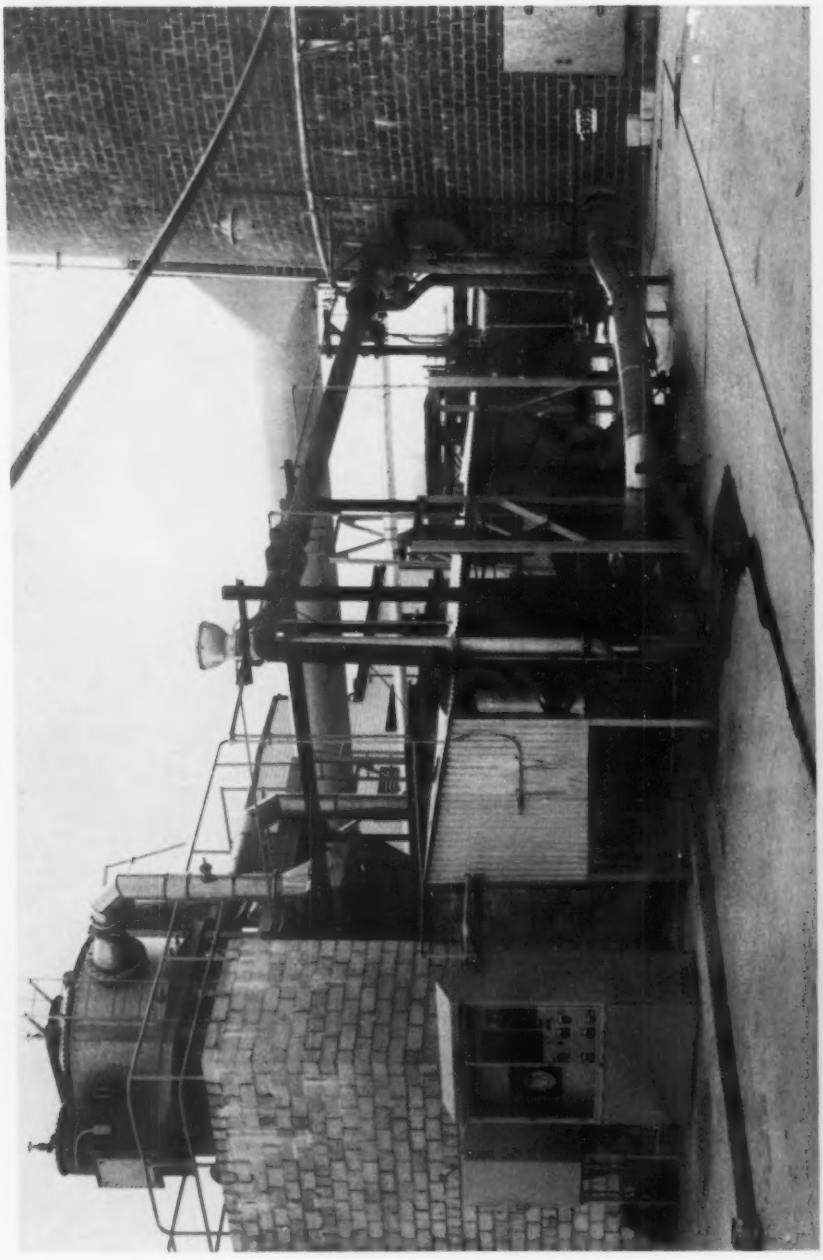

Fig. 9—Central Gas Disposal Area.

Fig. 10-Central Gas Disposal Area.

Fig. 9-Central Gas Disposal Area.

dust load was less than 900 grams per day of particle with a maximum diameter of 600 microns, a large number of them being in the submicron range. The designed efficiency of this house exceeded 99.9 per cent for particles down to 0.3 micron in size.

The roughing filters are standard commercial deep-pocket filters, each pocket containing two layers of 1/2-in.-thick graded submicron glass fibers packed to a density of 0.6 pound per cubic foot. The polishing filters are high efficiency paper units developed for the AEC. These filters have a particle removal efficiency of 99.5% at 0.3 micron particle size.

All access to the filters is through removable roof slabs which provide a method for remote maintenance. A canal located across the front of the filter house provides a receptacle for the dust-laden filters when the filter medium is being renewed. Precautions are taken to seal all filters in place in structural steel frames to ensure against leaks and bypassing of the filters.

Maintenance of the filter house is practically nonexistent except for periodic renewal of the filter media. The average life of the roughing filters is two years and of the polishing filters two and one-half years. The roughing filters are changed, one cell at a time, at approximately six-month intervals when the pressure drop across the filter house approaches or exceeds 8 inches water gauge. The filters are washed down, removed, and stored in the canal. The filter media are removed from the pockets and buried in the solid waste disposal area, and the pockets are reloaded for the next change.

The polishing filters are all changed at the same time when the pressure drop across them approaches 5 inches water gauge. To confine the dust, both sides of the filters are sprayed with strip coating before they are removed from the building for burial.

Homogeneous Reactor Test (HRT) Reactor Off-gas: The off-gas from the HRT is a 1/2 liter per minute stream of oxygen saturated with water vapor and containing small quantities of xenon and krypton isotopes. The water vapor is removed by cooling the gas to -30°F. The xenon and krypton isotopes are adsorbed and held on a bed of activated carbon until they have decayed to a very low activity level. The oxygen carrier gas is discharged to the stack.

Four carbon beds are installed in parallel at the HRT. Only two of these beds are expected to be used at any given time; one bed is a spare and the fourth bed serves the processing plant. Each bed consists of 40 ft of 1/2-in. pipe, 40 ft of 1-in. pipe, 40 ft of 2-in. pipe, and 60 ft of 6-in. pipe. The pipes for each bed are filled with about 410 pounds of activated charcoal pellets. The beds are submerged in water (for cooling) in an underground pit.

The off-gas from the reactor passes first through cold traps which freeze out the excess water vapor. The dry gas then passes through a long, empty three-inch pipe. The transit time through this pipe is about 7 hr, long enough for the most active isotopes to decay. The gas then goes to the carbon beds where the xenon and krypton are adsorbed on the carbon and held long enough (about 13 days for the krypton and 210 days for the xenon) for all but the krypton-85 activity

to decay away. The activity discharged to the stack is expected to be about 6 curies per day of krypton-85.

Oak Ridge Research Reactor (ORR) Off-gas: Off-gas from experiments connected with the ORR is discharged into the Laboratory vessel off-gas system, where it is cleaned before discharge to the atmosphere. This cleaning system has been described above.

The ORR building is equipped with a special air removal system for use in the event a catastrophe results in a release of fission products within the building. On activation by a monitoring system located within the ORR building, all doors, windows, etc., are closed; the normal building ventilation system is shut down; and a 5000 cfm emergency exhaust system is started. This emergency exhaust system is designed to maintain the ORR building under a negative pressure of 0.3 in. of water as long as required. This negative pressure ensures that all leakage of air through the building walls will be inward and will be exhausted under controlled conditions for cleaning prior to atmospheric discharge via a 250-ft stack located at the central gas disposal area. Cleaning of the exhaust air is accomplished by filtration for particle removal and scrubbing with caustic solution in a packed tower for iodine removal. Upon activation of the exhaust system, the caustic scrubbing system is started automatically.

IV. Current Handling and Disposal Methods for Solid Wastes; Stream No. 4 (Fig. 2)[11b]

Oak Ridge National Laboratory not only produces quantities of radioactive solid wastes but also serves as the major disposal site for such wastes from Argonne National Laboratory, Knolls Atomic Power Laboratory, Mound Laboratories, Battelle Memorial Institute, General Electric Company in Evansdale, Ohio, and other off-site installations. Radioactive solid wastes from these sources occupy approximately half the burial ground space currently being used at ORNL.

Nearly all the solid wastes routinely produced at ORNL consist of trash such as paper towels, glassware and other expendable items contaminated with radioactive materials. Occasionally contaminated concrete and wood and large pieces of equipment that cannot be economically salvaged by decontamination methods must be buried. Compared to the liquid wastes previously described, the solid wastes usually contain minor quantities of radioactivity, probably about 1% of the total disposed of by the Laboratory. Accurate determination of the curies handled is impossible, because the wastes are a nonuniform mixture of many kinds of materials.

In collecting contaminated solid wastes the prime considerations are personnel safety and preventing the spread of contamination. Ordinary galvanized garbage cans are used inside buildings as solid waste collectors. These "hot cans" are painted a distinctive yellow color and marked with a radiation symbol to distinguish them from waste cans used for uncontaminated trash. Hot cans are leak-proof and fitted with suitable covers to prevent the spread of contamination during handling. Frequent routine monitoring of the waste can areas is conducted to be sure that accumulation of materials does not produce radiation levels that will be hazardous. Special leak-proof Dempster Dumpsters are used to collect larger volume wastes. Placed at convenient locations outside certain buildings these are also distinctively marked for contaminated trash collection only.

Personnel who handle radioactive solid wastes are provided with protective clothing--cover-alls, shoes or boots and cloth or rubber covered gloves--to prevent contamination of personal clothing and body parts. Exposure above prescribed limits requires assignment to other work until the weekly average exposure falls below maximum permissible limits. Where an inhalation hazard exists, personnel are provided with filter-type Assault Masks. A radiation survey is made before moving materials to determine the radiation levels present and to establish working time for prevention of personnel exposure in excess of maximum permissible limits. Where radiation levels are sufficiently high to preclude handling the materials manually, remote handling by cranes or behind shields is practiced. An example of this type handling is the removal of "hot" dirt resulting from a liquid waste spill or break in a liquid waste line to the tank farm. All personnel are monitored with film badges and dosimeters to measure their radiation exposure.

Handling of wastes from outside agencies presents a number of problems and requires the use of a varied number and type of personnel. At one time the outside agencies were not required to identify the composition of materials shipped to ORNL as waste. However, after several incidents in handling unknown hazardous chemicals, the shippers have been required to describe their solid wastes in writing before they are accepted. Materials shipped by commercial

carrier must be packaged in such a way that they can be easily handled and so shielded that they will meet U. S. Interstate Commerce Commission regulations for shipment of radioactive materials. Wastes shipped in small lots by truck may require only one or two laborers to unload the materials and a health physicist to monitor the operation. These wastes are dumped directly from the truck into the burial pit.

Large shipments which arrive by rail are surveyed to determine the radiation levels involved before unloading and transfer to the burial ground. If broken containers are found, necessary precautions must be taken to prevent the spread of contamination during unloading and transit to the burial site.

After completion of the burial operation, men, equipment and vehicles are checked and decontaminated as necessary. This may even necessitate tearing out and replacing sections of the floor or other parts of a vehicle. Vehicles cannot be released to a public carrier for commercial transport of other materials until satisfactorily decontaminated.

Waste solids at ORNL are normally transported to the burial site by a group of special trucks used only for this purpose. These trucks are distinctively marked and fitted with leak-proof beds to prevent the spread of contamination during transit. A lead shield is placed between the can and the bed of the truck to provide protection for the crew against direct radiation. The pickup and loading of the materials are monitored by Health Physics personnel and the crew is advised as to the radiation levels involved and the maximum transit time allowable to prevent undue exposure.

Especially hot materials require special precautions. This may consist only of the addition of more shielding to the "hot" truck, but in some cases a crane is used to reduce the radiation hazard to the transport crew.

Radioactive solid wastes are currently buried in a Conasauga shale formation in the waste disposal area of Melton Valley near the highly active liquid chemical waste disposal pits (Fig. 5). Where weathered, the shale is soft and lends itself to easy excavation with power shovels or similar equipment. Normally the burial site is staffed during the work day with only a heavy equipment operator who digs and backfills the burial pits.

Alpha contaminated wastes and beta-gamma contaminated wastes are handled and buried separately. Alpha contaminated wastes are buried in holes approximately 15 by 15 feet deep. The waste materials are placed in the hole and covered by about a foot of earth. Approximately 8 inches of concrete is then poured over the earth in the hole. The concrete is followed by an additional two or more feet of earth to fill the hole to surface level. The use of concrete is a precaution to prevent anyone from inadvertently digging into the long-lived plutonium-contaminated alpha wastes at some future date.

Waste materials contaminated by beta-gamma activity from fission products are buried in trenches approximately 10 feet wide by 15 feet deep, which are dug completely across the burial site. Beginning at one end the wastes are dumped into the trench. When the first portion of the trench is filled, the wastes are

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covered by three or more feet of earth depending on the activity of the wastes. This dumping and back-filling procedure continues toward the other end of the trench until it is completely filled.

In special cases where very small, extremely active waste materials must be disposed of, auger holes approximately 1 foot in diameter and 15 feet deep are drilled in the burial site and the materials dropped into the hole. This procedure is used primarily to limit the radiation exposure to personnel during the back-filling operation.

The burial site is laid off in grid sections to facilitate efficient use of the entire area. Records are kept of the type and quantity of materials buried, and their locations are marked on the plot map.

Little information is available relative to the contamination of ground water by leaching or other means. However, the location of the site is such that the direction of ground water flow is toward White Oak Creek and the Clinch River into which low-level liquid wastes have been released almost since the beginning of the project. The burial site is completely fenced to prevent unauthorized entry and marked with radiation signs to warn personnel of the potential hazard.

Burial has been used at Oak Ridge National Laboratory as a means of disposal for solid wastes since the beginning of operations. To date four burial sites have been used. The earlier burial sites were relatively small and located without benefit of geological exploration. In two cases surface erosion was a problem, since the sites were located on the side of a hill. As each site was filled, it was abandoned. The site used prior to the present area was the largest of the earlier sites, comprising approximately 7 acres. It was used for a little over 5 years before being filled at an average annual burial rate of 1-1/2 acres. The current burial rate is of the order of 5 acres per year. This is an increase in the rate of burial site consumption by a factor of 3 in the last ten years. The increase may be attributed primarily to the increased use of ORNL burial facilities by outside agencies for the disposal of their solid wastes. At present, there are 25 or more outside agencies using these facilities. The present burial site is a relatively flat, cleared area of approximately 30 acres, with additional wooded land adjacent that can be used for expansion in future years.[16]

V. Surveys and Studies of Waste Disposal Effectiveness

The release of waste materials, either gaseous or liquid, to the environment places certain responsibilities upon the releasing agency. The releasing agency must safeguard the health and well-being of its employees; it must protect the public health and property in both the immediate and long-term aspects; and finally, it must assume responsibility to its own management with regard to legal implications associated with waste releases. In order to discharge such responsibilities, the Laboratory must necessarily control and evaluate its waste releases. The ORNL Health Physics Division performs this duty by maintaining, within prescribed limits, concentrations of radioactive wastes released by the Laboratory and by monitoring the surrounding area to prevent hazardous conditions that might arise from accumulations or reconcentrations of radioactive material. Several systems are employed to monitor for air-borne contamination and water contamination.

A. Monitoring for Air-borne Contamination

Atmospheric contamination and fall-out in the general environment are monitored by the following methods: (1) the three off-gas stacks in the X-10 area are monitored continuously for the purpose of preventing excessive discharges of radioactivity to the atmosphere for extended periods; (2) samples are collected at a number of stations by passing air continuously through filter paper from which air contamination may be evaluated; (3) fall-out is monitored by means of gum paper fall-out trays located at all sampling stations; (4) background measurements are made at many locations to determine the gradual increase in background radiation and to detect daily fluctuations in background. In addition, fall-out is monitored by collecting rain water samples.

Each air sampling method yields slightly different information relative to the nature of the air-borne contamination. Since the filter sampling equipment is located inside louvered enclosures, the heavier particles settle to the ground without being collected on the filters. This type sample contains only particles which might be considered of breathable size. The fall-out trays collect the heavier particles as well as the light particles and thus present the total fall-out picture. The background measurements indicate the radiation from contaminated soil and from radioactive gas such as natural radon or reactor-produced argon-41. Rain water samples contain both fall-out and "rain-out" and give information on the soluble and the insoluble fractions of the radioactive contamination.

1. Stack Monitoring

Continuous stack monitors are installed on the stacks at the Isotope Area, Hot Pilot Plant, and Graphite Reactor. The equipment at the isotope stack consists of a conventional LaPine motor blower unit which draws air from the stack gas stream through a millipore filter. At the Hot Pilot Plant stack the sample is drawn from the stack stream through a millipore filter by means of a steam jet arrangement. The sample at the Graphite Reactor is drawn from the duct downstream from the filter house and uses the pressure differential across the fan to draw the sample through Hollingsworth-Vose (H. V.) 70 filter paper. Mounted in a shielded box just below the H. V. 70 filter paper is a Geiger-Muller (G. M.) tube which monitors the activity collected on the filter. The counting rate is recorded on an Esterline Angus recorder.

The filters are changed daily at each stack and evaluated for gross beta and gross alpha activity. The data presented below represent samples covering a minimum of one month following installation. Average concentration values for gross beta and gross alpha activity and the highest average value for any 24-hour period from each stack are presented.

Concentration of Radioactivity in Stack Effluents

	Stacks		
	Hot Pilot Plant	Isotope Area	Graphite Reactor
Gross Beta			
Highest average per 24 hours	1.1×10^{-8} μc/cc	9.1×10^{-9} μc/cc	9.5×10^{-8} μc/cc
Average for period	3.2×10^{-9} μc/cc	9.3×10^{-10} μc/cc	3.9×10^{-9} μc/cc
Gross Alpha			
Highest average per 24 hours	1.5×10^{-10} μc/cc	5.5×10^{-11} μc/cc	1.3×10^{-12} μc/cc
Average for period	2.4×10^{-11} μc/cc	2.6×10^{-11} μc/cc	3.4×10^{-14} μc/cc

The average values listed in the table may be considered probable normal operating levels. When concentrations of activity in the stacks exceed these values, the values found must meet the test set forth by the following expression:

$$3\left(\frac{C_b \times 10^{-2}}{MPC_b} + \frac{C_a \times 10^{-2}}{MPC_a}\right) \leq 1$$

C_a = gross alpha concentration in stack effluent

C_b = gross beta concentration in stack effluent

MPC_b = breathing zone MPC (maximum permissible concentration) for gross beta activity

MPC_a = breathing zone MPC for gross alpha activity

The maximum ground concentration per unit discharge rate from each ORNL stack for varying weather conditions has been calculated by Myers and Purdy on the basis of Sutton's equations.[17] The data indicate a probable minimum dilution

factor of the order of 10^3 for each stack. Since a large segment of the Laboratory population will be involved in air-borne contamination from stack discharge, any uncontrolled exposure from this source should not exceed one-tenth the maximum permissible concentration (MPC). Consequently, the dilution factor of 10^3 has been reduced by a factor of ten with the result that 10^{-2} rather than 10^{-3} appears in the above equation. The factor of 3 is applied because there are three stacks which are monitored and evaluated separately. The factor of 3 in the equation obviates the possibility of each stack continuously discharging sufficient activity to create concentration of one-tenth MPC.

It should be emphasized that the above equation was developed to determine an action point for field investigation and subsequent corrective measures. If the left-hand side of the expression is greater than one, but less than ten, the condition is investigated. Identification of specific isotopes and consideration of their particular MPC values may reduce the quantity of the left-hand side to an acceptable value. If this is not the case, remedial action is taken. Should the value of the left-hand side exceed ten, remedial action is taken immediately and followed by an investigation.

2. Air Sampling and Fall-out Collection

a. Filter-type Air Monitors. Air samples are collected at all stations (local, perimeter and remote) by passing air continuously through H. V. 70 filter paper. At the local stations the activity collected on the filter is continuously monitored by means of a Geiger-Muller tube and logrithmic count rate meter, the readings being recorded on a Speedomax recorder (Fig. 11). In addition, the counting rates are telemetered via telephone lines to the area monitoring control laboratory where they are recorded on a twelve-point Speedomax recorder; each station reading is printed once every twelve minutes. The one-minute interval between station printings is necessary to compensate for the slow time constant inherent in the logarithmic rate meter circuit. The perimeter and remote stations contain no monitoring and recording equipment (Fig. 12). Under normal conditions of air contamination, the filters are removed once each week. When air contamination is significant, more frequent changes are necessary. The charts from the recorders in the local stations are changed normally at intervals of approximately three days. An hourly tabulation is made of the counting rate of the activity on each filter. Where air contamination in significant quantities is encountered and the half-life of the activity is long in comparison with the collection time, an indication of the concentration of activity in the air can be obtained from the rate of increase in activity on the filter. After removal, all filters are brought to the monitoring laboratory for analysis.

b. Gummed Paper Fall-out Trays. Radioactive fall-out is sampled by means of sedimentation trays or frames, each consisting of an aluminum frame (inside demensions 12 in. x 12 in.) to which is affixed a cellophane sheet gummed on one side (Fig. 13). The frame with the gummed paper is placed on a horizontal shelf attached to the air monitoring station. The gummed surface is exposed in an upward direction so that any particulate matter which falls upon this gummed surface becomes firmly fixed.

Fall-out samples are collected at all stations. Duplicate sets of fall-out frames are maintained; while one set is in the field for collection of particles,

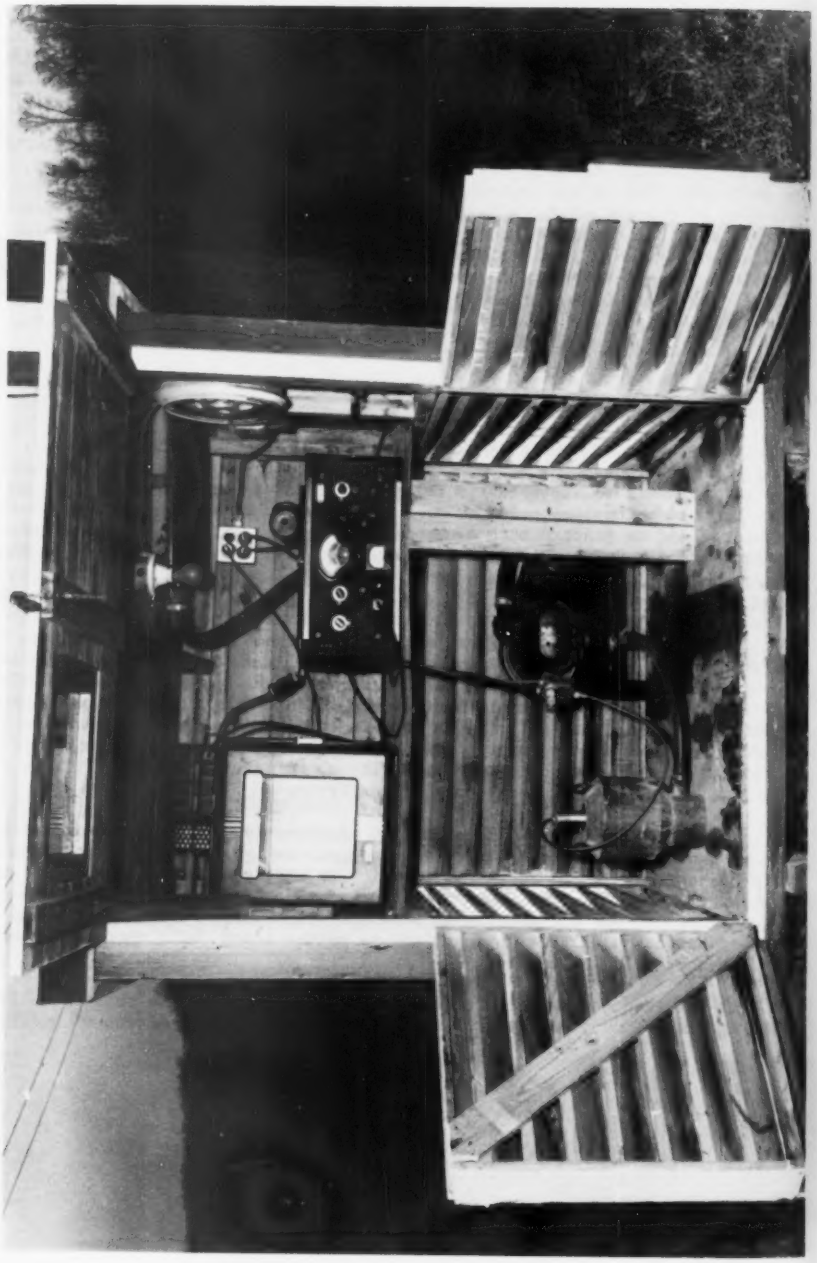

Fig. 11—ORNL Local Air Monitoring Station.

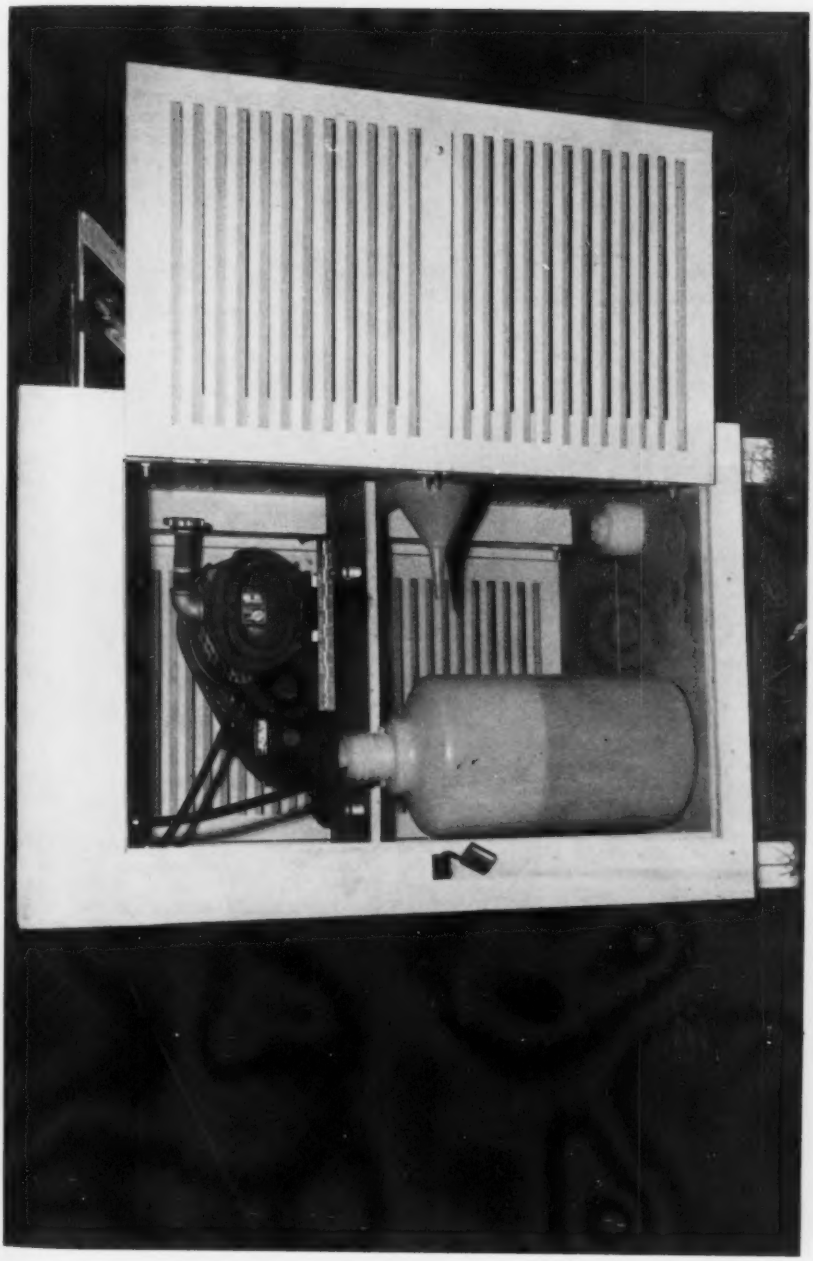

Fig. 12 - ORNL Perimeter and Remote Air Monitoring Station.

Fig. 11 - ORNL Local Air Monitoring Station.

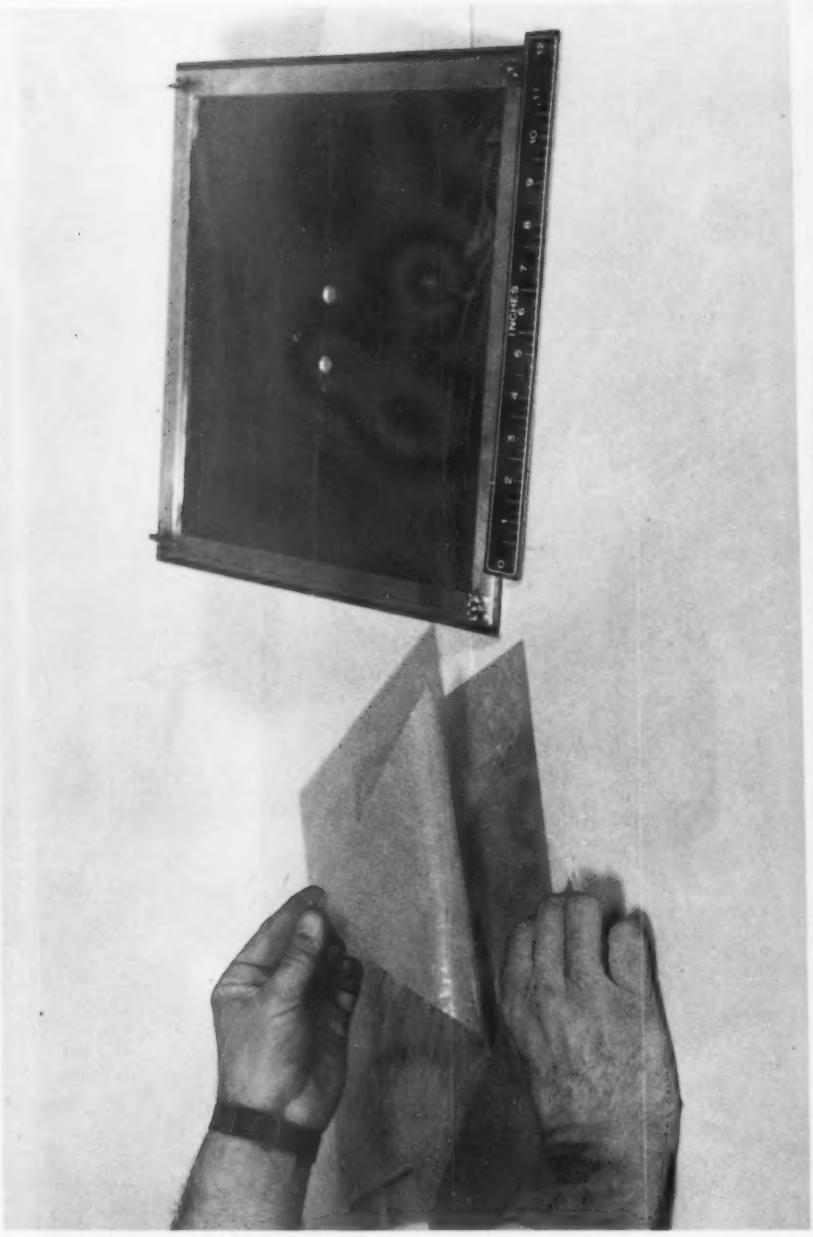

Fig. 13—ORNL Fall-out Sedimentation Tray.

the other set is in the laboratory for processing and analysis. Frames are exchanged normally once each week and transported in a special box which keeps them separated during transit.

c. Rain-out Samples. Rain-out measurements are made by collecting rainfall in 30-in. x 30-in. trays located at all perimeter and remote stations and one of the local stations. The collection tray serves as the roof of the monitoring station (Fig. 12). All rain collected is drained from the bottom of the tray into a polyethylene container inside the station. The depth of water in the container is measured to determine total rainfall from a calibration curve. Samples are collected normally once each week. The water in the container is shaken and one liter of the homogeneous mixture is taken to the monitoring laboratory for processing. The remainder of the water is discarded. The collection trays and the containers are cleaned at periodic intervals to prevent buildup of contamination from previous rains.

d. Analytical Methods. In the laboratory continuous air monitoring filters are mounted on 14-in. x 17-in. cardboards for autoradiographing. Twelve filters are mounted on each cardboard with the collecting surface of the filter facing away from the cardboard. The mounted filters are placed in the dark room in direct contact with a sheet of 14-in. x 17-in. Kodak Blue Brand X-ray film for an exposure time of 24 hours. The film and filters are weighted to assure complete contact during exposure.

The gummed papers are covered with a sheet of rubber hydrochloride, cut from the fall-out frame, and mounted on a 14-in. x 17-in. cardboard for autoradiographing, which follows the technique outlined above for continuous air monitoring filters.

With the use of a light table and the unaided eye, films are analyzed for radio-particulates by counting the number of particle images appearing on the films. In addition to counting the number of particles, some measure of each particle's activity is obtained by classifying each particle according to image diameter into one of several particular size ranges. The classification is accomplished by comparing image sizes with calibrated image sizes prepared from known particles. Four arbitrary order of magnitude size ranges are: $< 10^5$ d/24 hr,* $10^5 - 10^6$ d/24 hr, $10^6 - 10^7$ d/24 hr, $> 10^7$ d/24 hr. The results are tabulated for study and comparison with previous data and norms.

Following autoradiography, the continuous air monitoring filters are counted for gross beta activity to determine the average concentration of radioactivity in the air sampled. To evaluate the activity on the sample, the filter is affixed to the inside wall of a 1-in.-diameter aluminum cylinder, which is placed in a counter having a G. M. tube positioned along the axis of the cylinder. The sample is counted to determine the total activity in terms of μc/sample. The average air concentration in μc/cc is determined by dividing the total activity per sample by the volume of air sampled.

The gummed papers are each folded into a small square, placed in a crucible, and ashed in a muffle furnace. The ash is transferred to a stainless steel counting dish using a technique of transfer as follows:

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* d = disintegrations.

Fig. 13.-ORNL Fall-out Sedimentation Tray.

1. Spray the bottom of the counting dish with a wet film of acrylic resin.

2. Transfer the ash to the counting dish by sprinkling the ash onto the wet resin coating. Use a spatula to disperse the ash evenly over the dish.

3. Dry under a heat lamp. The resin bond, when thoroughly dry, provides a secure bond for retaining the ash.

After drying, the samples are counted in a gas-flow proportional counter to determine the gross beta activity. From these counts, the fall-out of gross beta activity in terms of μc/sq ft is determined.

Rain water samples are vacuum-filtered through Whatman 42 filter paper to remove the insoluble fraction of the activity. The filtrate is placed in a 1500-ml beaker and concentrated by evaporation on a hot plate to approximately 50 ml. The concentrate is then transferred to a 100-ml beaker (using nitric acid and distilled water) and evaporated to a few ml. This is transferred to a stainless steel counting dish and evaporated to dryness under a heat lamp. Care is taken to scrub down the walls of the beakers during each transfer. Each filter, containing the residue or insoluble fraction of the activity, is placed in a crucible and ashed in a muffle furnace. The ash is then transferred to a stainless steel counting dish using the method described for transfer of gummed paper ash.

All samples of both the soluble and insoluble fractions are counted in a gas-flow-type proportional counter to determine the gross beta activity. From these results, the concentration of activity in rain-out in terms of μc/cc is calculated.

When significantly high activity is encountered, an attempt is made to identify the specific radioisotopes and to determine their probable source. The samples are subjected to gamma spectrometry measurements and radiochemical analysis by the Analytical Chemistry Division to identify the radioactive constituents. In addition, the effective half-life and distribution of products in the sample are studied to differentiate between ORNL-originated activity and possible contamination from weapons tests.

e. Evaluation and Effectiveness of Air Sampling. From the air monitoring samples it is possible to determine the general level of air and soil contamination in the area, the source of specific contamination problems, and the improvement achieved over the years by better contamination control. Should the radioactive contamination reach a level that is a significant fraction of allowable limits, investigations into the source of the contamination are made and remedial actions taken immediately. Examples of such instances are the times when uranium oxide particles (and later protactinium particles) were found to be falling out on the X-10 area. Investigations led to the source of the particles, and the processes producing the particles were stopped until corrective measures had been completed. The remedial action required in the case of the uranium oxide particles was additional filtration of the off-gas system and a program for routine scanning of the Graphite Reactor for defective fuel elements. In addition, to prevent the particles already on the area from becoming air-borne and thus increasing the breathing hazard, the roofs and roadways in the area were washed and the barren areas within the X-10 area sown with grass.

An indication of the effectiveness of the air monitoring program for the X-10 area is the fact that the average air contamination value for 1957 was less than 0.1% of the maximum permissible concentration for continuous exposure. The maximum concentration measured was 170% of the maximum permissible concentration. However, this concentration continued for only a short period of time; and, therefore, the concentration averaged over a period of one week did not exceed the maximum permissible for that period.

To determine the air contamination in nearby populated areas, samples have been collected continuously at stations from 10 to 120 miles from ORNL. Although the activity levels have not been considered hazardous, the accelerated programs at ORNL necessitate continuing vigilance. The number and the efficiency of monitoring stations within an approximate 30-mile radius of the Laboratory are being increased.

f. Summary Data from Air Sampling and Fall-out Collection.

(1) Local Area. Average air contamination for 1957 in the X-10 area, determined from measurements made by ten continuous air monitors, was 0.1% of the MPC. The maximum weekly average measured by a single air monitor was 6.0% of the MPC. The maximum concentration measured was 170% of the MPC. In no period did the weekly average of activity concentration exceed the maximum permissible concentration. Table II shows the average and maximum contamination over the last 9 years. The weekly average number of radioactive particles per 1000 cu ft of air sampled was 4.2. The maximum number of radioactive particles for one week was 54.9. Table III shows the average and maximum number of particles over the past 9 years.

Gummed paper fall-out tray measurement gave a weekly average value for 1957 of 4.0×10^{-3} $\mu c/ft^2$. The highest total weekly fall-out for a single collection station was 8.4×10^{-2} $\mu c/ft^2$. The average number of radioactive particles per sq ft per week was 727. The maximum number of particles for any one week was 22,146.

The weighted average value for radioactive contamination in rain water, collected at one station only, was 7.6×10^{-7} $\mu c/cc$. The maximum concentration in any one week was 5.7×10^{-6} $\mu c/cc$.

(2) Immediate Environs (Within AEC Controlled Area). Average air contamination levels for 1957 in the immediate environs, determined by measurements at seven monitoring stations, was 0.02% of the MPC. The maximum weekly average for a single monitor was 2.39% of the MPC. The weekly average number of particles per 1000 cu ft of air sampled was 2.0 and the maximum number for one week was 20.2.

Gummed paper fall-out tray measurements gave a weekly average value of 9.2×10^{-4} $\mu c/ft^2$. The highest total weekly fall-out for a single collection station was 1.4×10^{-2} $\mu c/ft^2$. The average number of radioactive particles per sq ft was 563.9. The maximum number of particles for any one week was 25,848 per sq ft.

TABLE II

Air Activity Measured by Continuous Air Monitors

Units of 10^{-13} μc/cc

Year	Local Area		Perimeter Area		Remote Area		Year
	Max.	Average	Max.	Average	Max.	Average	
1949	2100.0	210.0					1949
1950	2040.0	180.0					1950
1951	1050.0	170.0					1951
1952	782.3	29.0					1952
1953	4407.3	46.4					1953
1954	203,183.5	478.4					1954
1955	2968.6	49.4					1955
1956	804.7	27.9	29.8	7.0	43.1	10.2	1956
1957	5607.0	82.3	229.7	14.8	47.3	9.0	1957

TABLE III

Particle Concentration Measured by Continuous Air Monitors

Particles Per 1000 cu ft

Year	Local Area		Perimeter Area		Remote Area	
	Max.	Average	Max.	Average	Max.	Average
1949	5.4	0.2				
1950	17.0	0.6				
1951	22.0	0.6				
1952	219.1	3.4				
1953	45.5	2.1				
1954	23.4	1.0			3.3	0.4
1955	13.0	1.4			29.2	1.0
1956	26.6	2.1	8.4	0.7	6.5	0.8
1957	54.9	4.2	16.3	2.0	15.8	1.9

The weighted average value for radioactive contamination in rain water was 8.1×10^{-7} μc/cc. The maximum concentration for one week was 5.1×10^{-6} μc/cc.

(3) Remote Areas (Outside AEC Controlled Area). At present, the Laboratory is operating one monitoring station outside the AEC controlled area approximately 120 miles from ORNL. The average contamination for 1957 measured by this station was 0.01% of MPC. The maximum for one week was 0.05% of MPC. The weekly average number of particles per 1000 cu ft of air sampled was 1.9 and the maximum number for one week was 15.8.

Gummed paper fall-out tray measurements gave a weekly average value of 4.7×10^{-4} μc/ft^2. The highest total weekly fall-out was 2.1×10^{-3} μc/ft^2. The average number of radioactive particles per sq ft was 69.2. The maximum number for one week was 2330.

The weighted average value for radioactive contamination in rain water was 8.1×10^{-7} μc/cc. The maximum for one week was 1.8×10^{-6} μc/cc.

3. Background Measurements

Contamination of the ground in the Laboratory area is monitored by means of background measurements at some 50 locations. The equipment used consists of a calibrated scaler, G. M. tube, and a portable 600-watt Fairbanks-Morse generator power supply. A pick-up truck is used to transport the equipment from station to station. With the equipment on the truck, the G. M. tube is mounted in a fixed position at the end of a 2-in. x 2-in. timber extending horizontally from the bed of the truck. A count is taken at each station and the results converted to mr/hr.*

Measurements indicate that the background level has increased only 0.005 mr/hr in the nearest residential community, Oak Ridge, Tennessee. The average level in the X-10 area during 1957 was 0.12 mr/hr or about 8 times the level 15 years ago (before the operation of the Laboratory); and in terms of present levels of acceptable exposure, this is not considered to represent a hazard. In fact, the background level from natural sources of radiation (e.g., granites, phosphates, etc.) is higher in many sections of the United States than the background in the X-10 area.

B. Liquid Waste Monitoring for Water Contamination

1. Stream Monitoring

a. ORNL Sampling Techniques. At ORNL the major volume of the liquid wastes are discharged via the settling basin to White Oak Creek. The discharge flow is measured by a 90° V-notch weir in conjunction with a standard stilling

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* The roentgen, r, is that quantity of x or gamma-radiation such that the associated corpuscular emission per 0.0001293 gram of air produces in air ions carrying 1 electrostatic unit quantity of electricity of either sign. One milliroentgen (mr) is 0.001 r.

well and Stevens water level or head recorder. The waste stream is sampled by a Trebler Proportional sampler equipped with a revolving dipper so shaped that individual dips collect a volume proportional to the flow through the weir at the time of each dipping. The sampler is equipped with a timer to limit the number of dips per hour, thereby limiting the sample to a volume suitable for handling and analysis.

Grab samples are collected daily at stream stations on Melton Branch and White Oak Creek (Fig. 1). These samples are used primarily to determine whether the activity detected is from the Laboratory proper or from the Melton Valley area (Section II, D). At White Oak Dam, located downstream from the confluence of White Oak Creek and Melton Branch (Fig. 1), continuous samples are collected for daily analysis.

b. Analytical Methods. Four types of samples are processed in the monitoring laboratory. First, those taken from the settling basin and White Oak Creek are evaluated for gamma activity to determine submersion exposure. Second, all of these are checked for gross beta activity. Third, composite samples of both settling basin and White Oak Creek effluent are analyzed radiochemically each week for plutonium content. Fourth, composite samples of both settling basin and White Oak Creek effluent are analyzed radiochemically each month for specific long-lived fission products.

Submersion exposure in terms of mr/hr gamma activity for settling basin and White Oak Creek is determined as follows:

Samples are put into 14-in.-diameter stainless steel containers to a depth of approximately 12 in. and counted by a calibrated G. M. tube enclosed in a water-tight brass cylinder placed along the axis of the container. The wall thickness of the brass tube around the G. M. tube is sufficient to cut out all beta activity. The counts obtained are corrected for background and the activity in mr/hr calculated.

The beta activity is determined by evaporating an aliquot of the sample to dryness in an aluminum counting dish and counting in a standard end-window counting set-up. The net counts per minute obtained are converted to 10% geometry and the beta activity in μc/cc calculated. The total curies of beta activity released is calculated from the flow measurements at the settling basin and White Oak Creek and from measurements of the specific concentration of activity in the effluent.

From the daily samples, a weekly and monthly composite is prepared for more detailed analysis for plutonium and long-lived fission products. The weekly composite samples are analyzed for plutonium to evaluate the alpha activity discharged. The monthly composite samples are concentrated and sent to the Analytical Chemistry Division for analysis for long-lived fission products by standard radiochemical procedures. The elements analyzed are strontium, cerium, ruthenium, iodine, zirconium, trivalent rare earths, niobium, barium, and cobalt,

From the analytical results, the fraction of the total beta activity comprised by each long-lived isotope is determined. A weighted MPC value for the mixture is calculated by the formula:

$$MPC_{mix} = \frac{\sum f_i}{\sum \frac{f_i}{MPC_i}}$$

where

f_i = fraction of the total beta each isotope represents

MPC_i = MPC value for each isotope

The calculated concentration in the Clinch River is compared to this weighted MPC value as a measure of the hazard existing.

c. Calculations. The submersion exposure to gamma radiation is calculated by multiplying the net counts per minute obtained with the calibrated G. M. tube by the appropriate conversion factor.[19]

In order to calculate the quantity of beta activity released, a knowledge of the flow and specific concentration from a given source is necessary. The specific concentration is obtained by analysis as previously described. The flow, in the case of the settling basin, is obtained from a continuous recording of head in the weir box and a rating table for the weir. Flow in White Oak Creek is obtained by summing the flows of White Oak Creek and Melton Branch, measured above White Oak Lake basin, and multiplying by a correction factor based on run-off areas to correct for the inflow between the gauging stations and outfall into the Clinch River. The flows in the two creeks are supplied by the U. S. Geological Survey, Surface Waters Branch, and are obtained from steam gauging measurements and continuous recordings of creek stage. The basic formula for determining the specific beta concentration of activity in the effluent is:

$$S_B = \frac{N}{2.22 \times 10^6}$$

where

S_B = gross beta concentration in $\mu c/cc$

N = net $d/m/cc$

2.22×10^6 = $d/m/\mu c$

The formula for the total curies of beta activity discharged is:

$$C_B = \frac{S_B \times F}{10^6}$$

where

F = total flow for the period in cc

10^6 = μc/curie

The concentration of activity in the Clinch River, which is the primary concern, is obtained by multiplying the specific concentration found in White Oak Creek by a dilution factor obtained from a ratio of the White Oak Creek flow to the Clinch River flow at the point of entry. This calculated concentration in the Clinch River assumes intimate mixing with White Oak Creek water upon entry of creek water into the river.

The average daily flow in the Clinch River at the point of entry of White Oak Creek is determined in the following manner. The 24-hour average (midnight to midnight) discharge flow from Norris Dan for the day in question and the preceding day are averaged and subtracted from the 24-hour average (midnight to midnight) Clinch flow obtained at the Scarboro gauge (Cl. 39.0) on the day in question to determine the local inflow below Norris. This figure is multiplied by 435/388, which is the ratio of the drainage area above White Oak Creek (Cl. 20.8) to that above Scarboro gauge to correct for side drainage below Scarboro gauge. To determine the flow at Cl. 20.8 this resulting amount is added to the average Norris discharge previously subtracted.

d. Clinch River Monitoring. Routine surveys of the Clinch and Tennessee Rivers are conducted to determine the extent of dispersion of radioactive material in river sediment, the levels of radiation encountered, and the consequent hazards to humans. Periodic measurements are required to predict the rate of buildup and, consequently, to determine the effectiveness of the liquid waste management program. Also, information is obtained relative to the effect on future industry if the radioactive content of bottom sediment in the Tennessee River systems is increased.

"Cross-section" measurements are taken every two miles in the Clinch River and approximately every 10 miles in the Tennessee River and its reservoirs (Fig. 14). A cross-section study consists of making radiation measurements of the bottom sediment activity and collecting sediment samples at predetermined intervals along the traverse from one bank to the other. Sampling points are located on TVA navigation charts and cross sections taken across the river at these points. Fifty-foot intervals are used in the Clinch River, but an average of ten readings and samples are taken per traverse in the Tennessee River and in Watts Bar and Chickamauga Reservoirs.

A device called a "flounder"[20] is used to measure the gamma radiation of the bottom sediments. The flounder consists of twelve battery-operated G. M. tubes connected in parallel. Pulses from the G. M. tubes are pre-amplified and recorded on a battery-operated scaler, the average count being determined by timing with a stop watch. Samples of bottom sediment for laboratory analysis are obtained with an Eckman dredge.

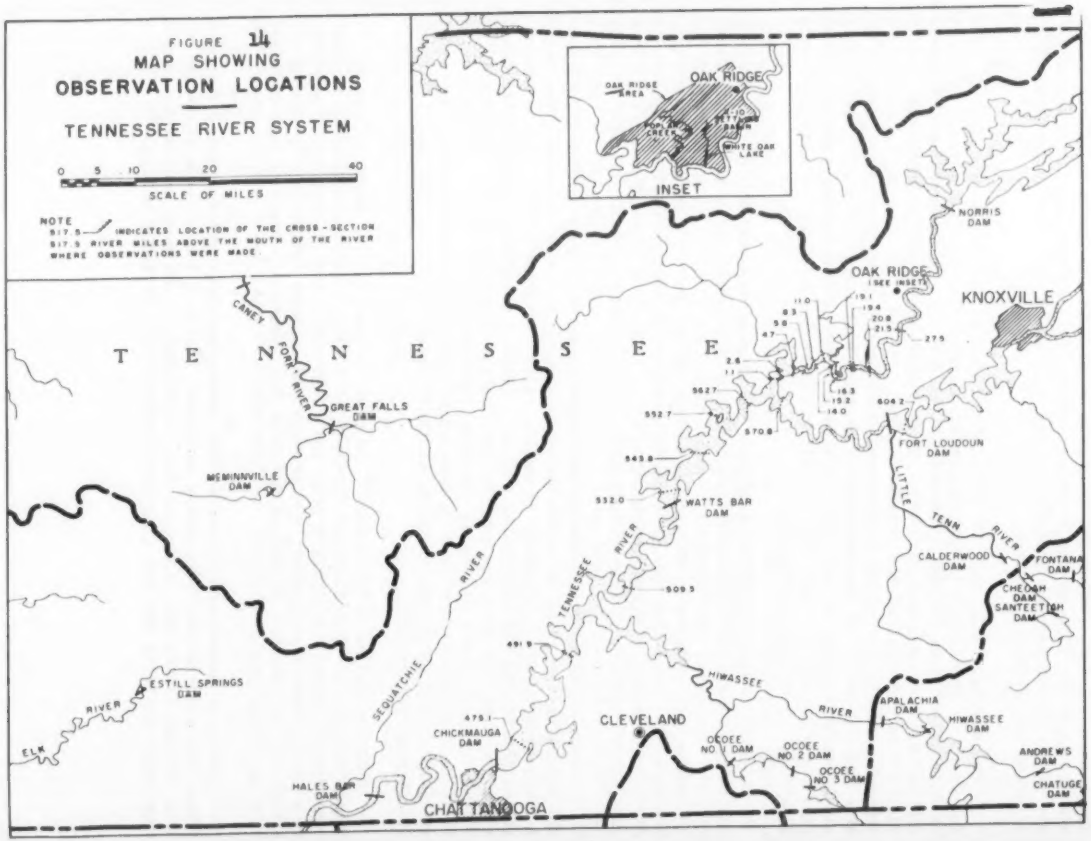
The gamma measurements made on the bottom sediment are corrected for background and averaged for each cross section. Plots of the average count vs. river mile for the Clinch and Tennessee Rivers are made. Sediment samples are radiochemically analyzed for the long-lived radioactive isotopes present.[21]

FIGURE 14
MAP SHOWING
OBSERVATION LOCATIONS

TENNESSEE RIVER SYSTEM

0 5 10 20 40
SCALE OF MILES

NOTE
517.5 — INDICATES LOCATION OF THE CROSS-SECTION
517.5 RIVER MILES ABOVE THE MOUTH OF THE RIVER WHERE OBSERVATIONS WERE MADE



A water sample is collected daily at Centers Ferry near Kingston, Tennessee, and composited for a three-month period. The composite is filtered and the filtrate concentrated. The residue or suspended solids and the concentrate are analyzed for fission products to determine the level and composition of the activity existing in the Clinch River.

From time to time, fish from the public waterway are obtained and analyzed for radioactivity. No significant levels of activity have been detected in these fish.[22] Samples of water from the river at the Oak Ridge Gaseous Diffusion Plant and at Kingston, Tennessee, the nearest population centers downstream, have indicated detectable amounts of fission products, but levels are well below those permitted for potable water. In fact, the drinking water for employees at the Gaseous Diffusion Plant is obtained from the river at a point approximately 8 miles below the outlet of White Oak Creek.

e. Summary Data. The total curies of fission products discharged from White Oak Creek to the Clinch River in 1957 was 397. The weekly average gross beta concentration in the Clinch River for 1957 was 0.9×10^{-7} μc/cc. The operating limit of 10^{-7} μc/cc gross beta activity in the Clinch River was exceeded 28% of the time during the year.* Table IV presents data on liquid waste released for the past 10-year period. Table V presents radiochemical analysis data for White Oak Creek effluent for the past 5-year period.

Radiochemical analysis of contaminants in the Clinch River shows activity for 1957 to be the following isotopes: TRE-28.8%, Ce-3.2%, Ru-15.0%, Zr-5.7%, Cs-22.4%, I-0.3%, Sr-20.9%, Cb-1.8%, Ba-0.4%, Co-1.2%.

The concentration of radioactivity in the sediment of the Clinch River drops off materially after the first 20 miles and approaches background levels at 150 miles. The maximum average activity for a single cross section is approximately 18 times background (approximately 0.1 mr/hr) and was found to be located at 12.8 miles below the outfall of White Oak Creek. At 100 miles downstream the level is approximately 2 times background and specific fission products are detectable by silt analysis.

2. Core Hole Monitoring

Evaluation of the effect of liquid waste with regard to the underground water table in the main X-10 area, in the burial ground, and in the area of the waste pit operation is made by means of core holes or monitoring wells. In the main X-10 area, measurements of underground activity detected by core hole monitoring reveal levels to be insignificant. In the waste pit and burial ground area, where most of the disposable high-level waste materials are sent, activity has been detected in the monitoring wells, but it has been established that the contaminant is primarily ruthenium-106, for which the maximum permissible concentration (MPC) in water is quite high, over a hundred times higher than the MPC for strontium-90. Several surface seeps have developed in the pit area, but the location of these areas is such that the direction of ground water flow is toward White Oak Creek and the Clinch River drainage basin into which low-level liquid wastes have been released and monitored since the beginning of the Laboratory.

* By calculation.

TABLE IV

Calculated Values of Radioactive Contamination Discharged to Streams

	Settling Basin Beta Curies Discharged		White Oak Creek Beta Curies Discharged		Probable Av. Concentration in Clinch River Units of 10^{-7} µc/cc		Plutonium Concentration Settling Basin Units of 10^{7} µc/cc		Plutonium Concentration White Oak Creek Units of 10^{-9} µc/cc	
Year	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.	Weekly Av.	Max.
1948	17.3	38.1	9.5	44.3	1.2	8.1				
1949	19.7	5.1	13.8	65.5	2.2	10.4				
1950	3.3	21.1	3.7	16.8	0.5	1.7	1.1	10.0	3.4	12.8
1951	3.3	20.0	2.0	6.8	0.3	1.7	1.1	6.3	4.7	42.1
1952	7.5	35.6	4.1	16.9	0.6	2.5	1.5	10.1	4.2	19.0
1953	5.6	34.8	5.8	27.5	0.9	6.3	1.5	19.7	8.5	112.4
1954	4.5	41.8	7.4	31.0	1.7	9.4	2.0	22.5	6.9	37.0
1955	4.1	16.6	8.4	53.6	1.3	12.7	2.5	36.1	18.6	173.3
1956	4.9	23.3	11.2	66.4	1.5	9.3	1.9	12.6	27.8	151.2
1957	3.6	13.3	7.6	22.4	0.9	3.0	1.0	3.7	12.2	35.5

TABLE V

Yearly Discharges of Radioactivity to Clinch River, 1950-1957

Year	Gross Beta[a] Curies	Per Cent of Gross Beta Activity Identified With Specific Radionuclides[b]										MPCw x 0.10 (Calculated)[d]	Probable Av. of Discharge[e]
		TRE[c] (-Ce)	Ce	Ru	Zr	Cs	I	Sr	Nb	Ba	Co		
1950	191												
1951	101												
1952	214												
1953	304	37.4	2.2	8.7	2.5	2.1	0.7	44.3	1.2	0.9	---		
1954	384	42.6	6.2	2.9	3.6	5.7	0.9	35.2	2.4	0.6	---	3.1×10^{-7}	1.7×10^{-7}
1955	437	33.7	19.4	7.0	1.2	14.4	1.6	21.1	1.3	0.2	1.5	4.4×10^{-7}	1.3×10^{-7}
1956	582	23.8	10.1	5.0	2.0	29.6	0.6	17.9	2.6	0.5	7.9	6.3×10^{-7}	1.5×10^{-7}
1957	397	28.8	3.2	15.0	5.7	22.4	0.3	20.9	1.8	0.4	1.2	5.3×10^{-7}	0.9×10^{-7}

[a] Gross beta activity in daily samples from Creek at White Oak Dam.

[b] Daily samples at Dam composited for monthly radiochemical separations and analysis.

[c] Trivalent rare earths exclusive of cerium.

[d] Maximum permissible concentrations for drinking water are calculated on the basis of the percentage of the specific radionuclides identified. MPC values for occupational exposure are determined and reduced by a factor of ten as recommended by the ICRP for drinking water for the population near a nuclear installation.

[e] Daily volumes of flow are determined from records of USGS gaging stations on White Oak Creek and Clinch River. The probable average concentration in Clinch River, as calculated, is the observed concentration in the Creek multiplied by the dilution factor (i.e. $\frac{\text{Flow in Creek}}{\text{Flow in River}}$) averaged over the same period of time - daily, weekly, monthly, or yearly.

C. Conclusions

In conclusion, Oak Ridge National Laboratory is contributing some radioactive materials to the local environment, and continuing vigilance must be maintained to determine the nature and extent of dispersion of these radioactive materials. Up to the present, data collected from the current monitoring systems indicate that the activity released is below the maximum permissible levels recommended by the National Committee on Radiation Protection and the International Commission on Radiological Protection. The low degree of radioactive contamination of the air and water by ORNL does not represent a hazard to the local environment or population.

VI. Acknowledgement

The authors express grateful appreciation to the following co-workers for information and assistance in the preparation of this report: F. L. Culler, Jr., W. K. Eister, K. Z. Morgan, and E. G. Struxness of Oak Ridge National Laboratory and O. W. Kochtitzky of the Tennessee Valley Authority.

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Frank N. Browder, born October 29, 1920, in Montgomery, Alabama; B.S. in Chemical Engineering 1942 from Alabama Polytechnic Institute, Auburn, Alabama; chemical engineer in production and process improvement work at Huntsville Arsenal, Alabama, and Rohm and Haas Company, Knoxville, Tennessee, from 1942-1945; from 1945 to present served in research and development engineering in atomic energy field at Clinton Engineer Works - Tennessee Eastman Corporation (Y-12) and at Oak Ridge National Laboratory (X-10). From 1948 to present served as design engineer for ORNL and Idaho Chemical Processing Plant radioactive waste disposal systems.

STATEMENT ON WASTE MANAGEMENT AT THE OAK RIDGE AND PADUCAH PRODUCTION PLANTS

Logan B. Emlet, Manager of Production
Union Carbide Nuclear Company
Division of Union Carbide Corporation (see p. 523)

The purpose of this paper is to describe the manner in which radioactive waste management problems are handled at the production plants which Union Carbide Nuclear Company operates for the Atomic Energy Commission, radioactive waste being defined as comprising two categories: (1) contaminated material containing small quantities of uranium for which the cost of recovery would exceed the value of the material obtained, and (2) production tails; that is, relatively pure uranium compounds depleted in U-235. The term "disposal" includes (1) release to the environment, and (2) commission to storage for an indefinite period.

The procedures to be discussed are those involved in the collection, handling, processing, disposal and monitoring of the radioactive wastes generated. These wastes are of three types: liquids, solids, and gases. The liquids consist of mixtures of solutions of various types but primarily of raffinate, spent chemical solutions, and very dilute aqueous solutions originating from equipment cleaning operations. The solids are more diverse in nature, ranging from contaminated paper and floor sweeping compound to relatively pure material, depleted in U-235, which is committed to storage as hexafluoride and other forms. The radioactive waste gases are principally dilute uranium hexafluoride in stack gases, and dusts or mists from machining and chemical operations.

There are two principal considerations which influence the quantities of radioactive material released to the environment; namely, economy and safety, economy being the guiding principal provided that governing health and safety regulations are met.

The safety standards used are those published by the National Committee on Radiation Protection and adopted by the AEC (Manual Chapter 0550), establishing the maximum levels of radioactivity permissible in the atmosphere and streams affected by the operations, and on surplus and scrap materials released. It is our aim to combine the interests of economy and safety in such a way that the cost of our product is minimized and the environment both inside and outside the plant area is not impaired.

Existing operational methods and procedures are described separately for the three production facilities operated by our Company.

Waste Disposal at the Oak Ridge Gaseous Diffusion Plant

Waste disposal operations at the Oak Ridge Gaseous Diffusion Plant are concerned exclusively with uranium. The problem does not involve highly radioactive waste materials such as those associated with reactor operations. Most of the discard flows at this plant originate from process equipment cleaning operations. In addition to this, small quantities escape to the atmosphere in the form of uranium hexafluoride.

A map of the Oak Ridge Gaseous Diffusion Plant area is appended to show the location of operations which release wastes to the environment and locations where monitoring operations are performed. Those roads open to the public are indicated; numerous auxiliary roads restricted to authorized personnel are not shown.

Liquid Waste

The decontamination facilities for process equipment cleaning are designed to remove surface contamination through nitric acid scrubbing. This acid solution is then stripped of uranium through a sequence of extraction steps. The acid is reused in the decontamination step while the uranium is converted to oxide form for ultimate return to production cycles. This process operates essentially on a continuous basis. Therefore, the waste material flows associated with the process originate and are discarded on essentially a continuous basis. These discard flows are in the form of aqueous extraction column tailings (raffinate), condensate, spent acid, and rinse water.

Each of these discard flows is monitored routinely for uranium content. This means that a liquid sample representative of each discard batch is withdrawn for laboratory analysis. The frequency of discard batch generation is dependent upon operational needs, but, under normal conditions, will be in the order of 15 to 20 batches per day for all discard flows. Evidence of excessive uranium is defined as that quantity having an economic value greater than the cost of further recovery processing. In view of the dollar value associated with this material, particularly when enriched in the isotope 235, very small relative quantities become excessive. As the solutions become sufficiently depleted, they are discharged batchwise to the settling basin. The route of these flows is noted as Point A on the ORGDP map. The settling basin is an earth retained, open pit body of water approximately 50 yards wide by 100 yards long and averages about four feet in depth. The overflow water from the settling basin discharges to an off area flowing stream (Poplar Creek) which, in turn, empties into the Clinch River on the west side of the area. Downstream from the plant premises, the alpha activity of the water in the Clinch River is approximately 3.5×10^{-9} microcuries per milliliter, the range being about 2 to 5×10^{-9} microcuries per milliliter. This concentration is based upon samples withdrawn from the flow channel approximately one mile below the discharge point of Poplar Creek. Background

measurements for the Clinch River indicate an alpha activity of about 7 x 10^{-13} microcuries per milliliter. It is to be noted that the downstream uranium contamination in the Clinch River is well below the maximum allowable concentration of 7 x 10^{-6} microcuries per milliliter. This concentration is based on limits recommended by the National Committee on Radiation Protection (AEC Manual, Chapter 0524). Future discards are expected to be in the order of 17 millicuries per year. This activity will be contained in about 3,000,000 liters of solution which will pass through the settling basin before entering the off-area effluence. No increase in the present uranium burden of the Clinch River is expected. The alpha activity of the discard flows is expected to be about 5.5 x 10^{-6} microcuries per milliliter, which is below the maximum allowable concentration without the dilution effect of Poplar Creek.

Gaseous Waste

A second category falling within the definition of radioactive waste involves uranium hexafluoride. This effluence occurs as a discharge to the atmosphere at the location shown as point B on the ORGDP map, and originates from the conversion of uranium tetrafluoride to the hexafluoride form. The feed material for gaseous diffusion plants being uranium hexafluoride, uranium is processed to this form from oxides in feed manufacturing facilities. In the last processing step, the product, in the gas phase, is passed through a series of refrigerated traps where virtually all of the material is condensed. Because there are also noncondensable gases, such as nitrogen and oxygen, in this stream, it is necessary to maintain a vent to the atmosphere. A small amount of uranium hexafluoride escapes in this venting operation. These vent gases are sampled at eight-hour intervals to determine the amount of uranium passing to the atmosphere. Experience has indicated this flow to be approximately 0.7 millicurie per day.

Air samples within the plant control area are obtained over an eight-hour period every eighth day, at a fixed point indicated on the ORGDP map. The sampling station is located away from the effects of large buildings on air currents and near the downwind direction from the principal point of discharge of uranium compounds to the atmosphere. The average concentration found, 5 x 10^{-13} microcuries per milliliter, is below the maximum permissible concentration for use beyond the control area, 1.7 x 10^{-12} microcuries per milliliter, the limit recommended by the National Committee on Radiation Protection (AEC Manual, Chapter 0524).

Solid and Sundry Waste

In the course of diffusion plant operations, a great many commonly used materials become contaminated with uranium. These may be lubricants, such as hydrocarbon and fluorocarbon oils, absorbent paper, activated alumina,

and both metal and fiberboard containers. Because of the nature of the operation, there is generally a backlog of contaminated materials awaiting recovery. These materials consist of both solids and solutions. As such, these are not considered waste material, however, because some recovery can be accomplished. For example, combustibles are burned in specially adapted incinerators with the uranium remaining in the ash. Activated alumina is dissolved in nitric acid and metal containers are washed, the resultant solutions being passed through the recovery equipment. These instances are mentioned to point up the fact that while uranium may appear in many unusable forms, the ultimate discard to the environs, being a function of the decontamination and recovery process, is in the form of aqueous extraction column tailings, condensate, spent acid, or rinse water.

Part of the uranium oxides processed at ORGDP consists of material recovered from plutonium-producing reactors and contains minute quantities or fission products and plutonium. These constituents are retained in the feed plant ash. The major portion of the ash is stored in iron drums. One shipment was made to Oak Ridge National Laboratory for further recovery.

Instances occur when the contamination level of unserviceable equipment is excessive or when contaminated solid materials are accumulated which do not warrant recovery processing from an economic viewpoint. In these cases, the equipment or material is buried at the ORNL facility.

Unserviceable or surplus equipment having resale value is sold on an "as is" basis if security and contamination restrictions are met. In certain instances smelting may be performed to achieve decontamination. Equipment to be sold must meet the contamination limits set forth by AEC regulations (AEC Manual, Chapter 5170-1411, "Utilization and Disposal of Excess and Surplus Personal Property"). Material to be disposed by sale as scrap must meet the AEC regulations on scrap (AEC Manual, Chapter 5182-05, "Disposal of Surface Uranium-Contaminated Metal Scrap").

Waste Disposal at the Paducah Plant

The principal functions of the Paducah Plant are similar to those of the Oak Ridge Gaseous Diffusion Plant. It follows that the majority of wastes are similar in nature. The exceptions, for example solid waste from the Metals Plant at Paducah, are described in the text which follows.

A map of the Paducah Plant and the immediately surrounding area is appended, showing the tributaries to the Ohio River which receive the liquid effluents from the plant, and the locations where monitoring of the drainage water is performed. The solids waste pit and cylinder storage yard for uranium hexafluoride are also shown.

Liquid Waste

Effluents containing radioactive waste materials come primarily from three sources: the Decontamination and Recovery Building, the Feed Plant, and the Metals Plant. Filtrate from uranium precipitation processes, aqueous extraction column tailings, and miscellaneous sources of uranium-contaminated water are conducted directly into one of two creeks which lead to the Ohio River. The radioactive waste effluents from the Decontamination and Recovery Building and the Feed Plant are directed to a branch of the Little Bayou Creek via an open diversion channel. The uranium-bearing effluent from the Metals Plant enters the Little Bayou Creek by a pipe line directly east of this facility. The Little Bayou as well as the Big Bayou Creek carries other Plant waste waters in addition to run-off from a large area as can be seen on the map.

The Big and Little Bayou Creeks are monitored for uranium content at points marked A, B, and C. Samples are obtained weekly at points A and C and monthly at point B. Average uranium concentrations in activity units obtained from point A are 5×10^{-8} microcuries per milliliter; point B, 1.1×10^{-8} microcuries per milliliter; and point C, 1.2×10^{-7} microcuries per milliliter. These are well below the maximum permissible concentration recommended by the National Committee on Radiation Protection which is 7×10^{-6} microcuries per milliliter beyond the control area (AEC Manual, Chapter 0524).

Part of the feed material at the Paducah Plant consists of uranium compounds recovered from plutonium-producing reactors, and contains minute quantities of residual fission products and plutonium in concentrations so low that the oxides from this source can be handled as any other uranium oxide. These contaminants are separated from the uranium in the uranium hexafluoride manufacturing plant and eventually, in the course of recovery and decontamination processes, are discarded in the aqueous waste streams from the Plant. The concentration of fission products and plutonium in the creek carrying these wastes is well within the maximum permissible concentration recommended by the National Committee on Radiation Protection (AEC Manual, Chapter 0524).

Gaseous Waste

There are two vent stacks in the Plant which vent systems handling uranium hexafluoride. These ventsdischarge approximately 1.75 millicuries per day of uranium hexafluoride to the atmosphere in the course of Plant operations. The major source of discharge occurs in the processing of uranium tetrafluoride to uranium hexafluoride. The uranium hexafluoride is removed by condensation in refrigerated traps for subsequent transfer to shipping containers. A minute portion, however, does not condense but is carried through the refrigerated traps with a flow of lighter gases such as nitrogen and oxygen. These gases discharge through a fluid bed absorber prior to being vented to the atmosphere. The other source of discharge occurs from the inclusion of minute traces of uranium hexafluoride in the venting of light

gases from the top of the gaseous diffusion cascade. These light gases are inadvertent diluents such as air and are separated from the uranium hexafluoride in the cascade and vented to the atmosphere.

While 1.75 millicuries per day of uranium hexafluoride is extremely small and represents no health hazard, it is our policy to continuously strive to reduce the uranium losses to negligible monetary losses. A more efficient absorber unit is presently being designed to further reduce the discharge of uranium hexafluoride to the atmosphere to approximately 0.25 millicuries per day.

Monitoring of the air inside the control Plant area has repeatedly shown no contamination above background by alpha counting continuous samples taken over an 8-hour period. Also an air sampling program was conducted to obtain samples around the clock inside the control area for photofluorimetric determination. The highest value found was 1×10^{-13} microcuries per milliliter of air. This is well below the maximum permissible concentration recommended by the National Committee on Radiation Protection which is 1.7×10^{-12} microcuries per milliliter beyond the control area (AEC Manual, Chapter 0524).

Solid Waste

Solid radioactive waste materials from which it is presently uneconomical to recover the uranium are being stored in an open pit that was formerly used as a holding pond for waste solutions. The pond was drained of its liquid content and now provides approximately 350,000 cubic feet of open storage space for solid waste materials from the Metals Plant. The major types of waste material containing uranium are crucible "burn-out" from the casting operation, sawdust from the ingot cutting operation, vacuum cleanings from various areas and magnesium fluoride slag from the metal reduction operation. The uranium contained in these solid wastes presently has no value and there are no plans for recovery. Since the uranium metal and compounds contained in these waste materials are relatively insoluble and since the subsoil is a dense clay, the uranium will not leach away into ground water. Periodically, the pond will be drained of accumulated rain water to one of the creeks. A recent analysis of the water in the pond indicated 1.3×10^{-6} microcuries per milliliter.

Ash or residue from the manufacture of uranium hexafluoride is sometimes stored temporarily in drums for recycle during periods of low feed availability. This type of activity is not considered a waste disposal problem as previously defined.

The quantities of uranium hexafluoride from which the U-235 has been stripped is stored in cylinders. This material will be stored indefinitely as the hexafluoride and other forms.

Waste Disposal at the Y-12 Plant

At the Y-12 Plant wastes are generated during the fabrication of nuclear weapon components and in the process of recovery of uranium from a wide variety of materials for other Commission contractors. The many and diverse operations involved generate a number of different types of wastes which may be either liquid, solid, or air-borne materials. These wastes are such that it is not necessary to process them merely for reduction of volume or level of radioactivity prior to storage or disposal. The one exception to this is some used equipment or materials which may require decontamination to permit their sale to the general public as salvage or excess.

A map of the plant area is appended showing the locations where wastes are released to the environment and the locations where monitoring operations are performed.

Liquid Wastes

Liquid wastes consist of solutions containing small concentrations of uranium, the recovery of which is not economically feasible. These wastes may be spent process solutions -- raffinates, supernatants, pickling and plating solutions; or they may be wastes from decontamination or cleaning operations, housekeeping procedures, etc.

Liquid wastes are pumped or drained to the disposal areas via pipelines. In some temporary situations they may be transported by tank truck. The major portions of the wastes are carried via the process sewers and released into the East Fork of Poplar Creek. The alpha activity of these liquids is less than 6×10^{-4} microcuries per milliliter before dilution and less than 4.5×10^{-7} microcuries per milliliter after dilution in the stream within the control area. It is noted that the East Fork of Poplar Creek joins another stream to form Poplar Creek which then passes through the ORGDP area and empties into the Clinch River. Thus, activity measurements of water samples downstream from ORGDP reflect the release of wastes from both ORGDP and the Y-12 Plant.

The more concentrated chemical or acid solutions, such as spent, pickling, or plating solutions, are disposed of via pipeline to settling basins west of the Y-12 Plant area. The earthen dikes of these basins permit no above-ground release of the impounded wastes or influx of ground waters. Evaporation provides adequate control of the liquid level. Alpha activity of the impounded liquids is less than 7.2×10^{-2} microcuries per milliliter.

The two streams which drain the Y-12 valley, Bear Creek and the East Fork of Poplar Creek, are sampled to monitor the release of radioactive liquid wastes to the environment beyond the control area. Sampling points are

indicated on the appended map. The sample from the East Fork of Poplar Creek is a weekly composite of daily samples. This stream, which is the larger of the two and traverses the length of the plant area, carries the major portion of the Y-12 Plant liquid wastes. The sample from Bear Creek is a once a week spot sample. Bear Creek rises to the west of the plant area and flows west. Its content of radioactivity possibly is suggestive of the gradual release of contamination from the settling basins, burning ground, and burial ground which are to the west of the Y-12 Plant. Samples from both of these streams are analyzed for both alpha and beta emitting radioactive materials. The levels of activity in these samples have been consistently less than the maximum permissible concentrations recommended by the National Committee on Radiation Protection and Measurement for water beyond the control area.

Solid Wastes

Solid wastes consist of contaminated used processing equipment and materials, contaminated trash and process residues, all of which may contain uranium in concentrations so small that further attempts at recovery are uneconomical. Solid wastes are routed to contaminated material storage areas, the decontamination and smelting area, Y-12 burning ground, the Y-12 burial ground, or the ORNL burial ground.

Contaminated combustible solid wastes having an alpha activity less than 10,000 disintegrations per minute per 100 square centimeters of surface area are burned in the Y-12 burning ground. Experience has shown that the burning of materials contaminated to this extent does not generate significant air-borne radioactivity. Combustibles, which are contaminated above this level, are buried.

Contaminated solid wastes having alpha activity less than 7×10^{-2} microcuries per gram are buried at the Y-12 burial ground. Wastes known to be contaminated with plutonium or thorium and those wastes having alpha activity greater than 7×10^{-2} microcuries per gram are sent to the ORNL burial ground.

Used contaminated equipment and/or materials are sold to the general public through property sales outlets. Items released to the public in this manner are considered to be excess or surplus and must fall within the contamination limits set forth in AEC Manual Chapters. Equipment and materials may require cleaning or decontamination to comply with the limits of subsection 5170-1411. Those items which cannot be decontaminated to within these limits may be either retained in contaminated storage areas or disposed of in one of the burial grounds. Certain types of metal scrap are released to the public if they meet the contamination limit requirements of Manual Chapter 5182-05. Some scrap metals contaminated with uranium in excess of the above limits are decontaminated by smelting as set forth in AEC Manual Chapter 5182-052. Slag from the

smelting operation is the final waste which contains the uranium contamination; it is buried in the ORNL burial ground.

Air-Borne Wastes

The air-borne materials released to the atmosphere within the Y-12 Plant are principally the uranium-containing dusts and mists from mechanical and chemical operations with small amounts of gaseous uranium hexafluoride from reduction operations. These atmospheric contaminants are actually minute concentrations of process material in off-gases after filtration and air cleaning equipment rather than wastes which must be disposed of. Sampling of these off-gases is performed at the several stacks to help estimate and control material loss.

Three outdoor air sampling stations are operated 24 hours per day, seven days per week, to monitor the radioactivity content of the atmosphere within the Y-12 Plant area. The location of these three samplers is noted on the appended map. The prevailing winds of the area blow from the uranium processing areas toward these three stations. The level of radioactivity in samples collected at these stations has been consistently less than the maximum permissible concentration of uranium in atmospheres beyond the control area (1.7×10^{-12} microcuries per milliliter), the limit recommended by the National Committee on Radiation Protection and Measurement.

Logan Bond Emlet was born on February 8, 1913, at New Oxford, Pennsylvania. He attended Dickinson College at Carlisle, Pennsylvania, where in 1935 he received a Bachelor of Science Degree in the field of Chemistry. He then attended the University of Pennsylvania, where he received a Master of Science Certificate in Chemistry in 1936.

Mr. Emlet was employed by E. I. duPont de Nemours & Company from 1935 to 1946. During his employment he held various positions in production supervision at duPont installations in Wilmington, Delaware; Pennsville, New Jersey; Gibbstown, New Jersey; Kankakee, Illinois; Chicago, Illinois; and Oak Ridge, Tennessee.

Mr. Emlet's work in the field of atomic energy began with his assignment by the duPont Company to the University of Chicago Metallurgical Laboratory in January, 1943. From there he was transferred to the Oak Ridge National Laboratory at Oak Ridge, Tennessee, and in 1946 he resigned his position with the duPont Company in order to remain in atomic energy work with the Monsanto Chemical Company at Oak Ridge. He was employed by Monsanto from 1946 until 1948, when the Union Carbide Nuclear Company assumed operation of the Oak Ridge National Laboratory. Mr. Emlet has continued in atomic energy work with Union Carbide since that time.

During his 16 years in atomic energy work, Mr. Emlet has held the following positions:

1943 - 1948:	**Supervision of operation of the Graphite Reactor, Oak Ridge National Laboratory, Oak Ridge, Tennessee**
1948 - 1950:	**Operations Division Superintendent, ORNL (Union Carbide)**
1950 - 1952:	**Executive Director, ORNL (Union Carbide)**
1952 - 1953:	**Superintendent, Y-12 Production Plant (Union Carbide)**
1954 - Present:	**Manager of Production (for the three atomic energy production plants operated by Union Carbide Nuclear Company at Oak Ridge, Tennessee, and Paducah, Kentucky.)**

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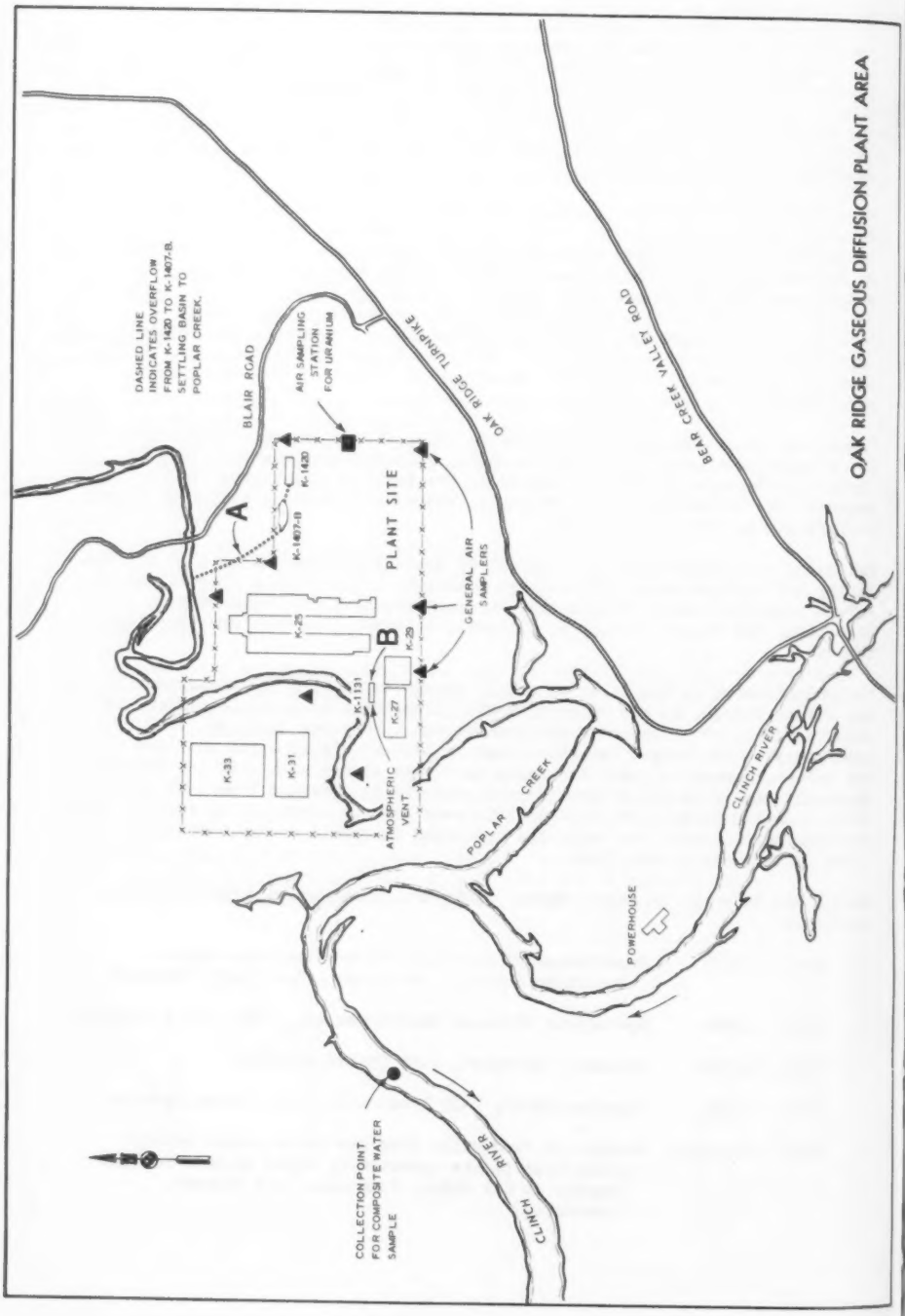
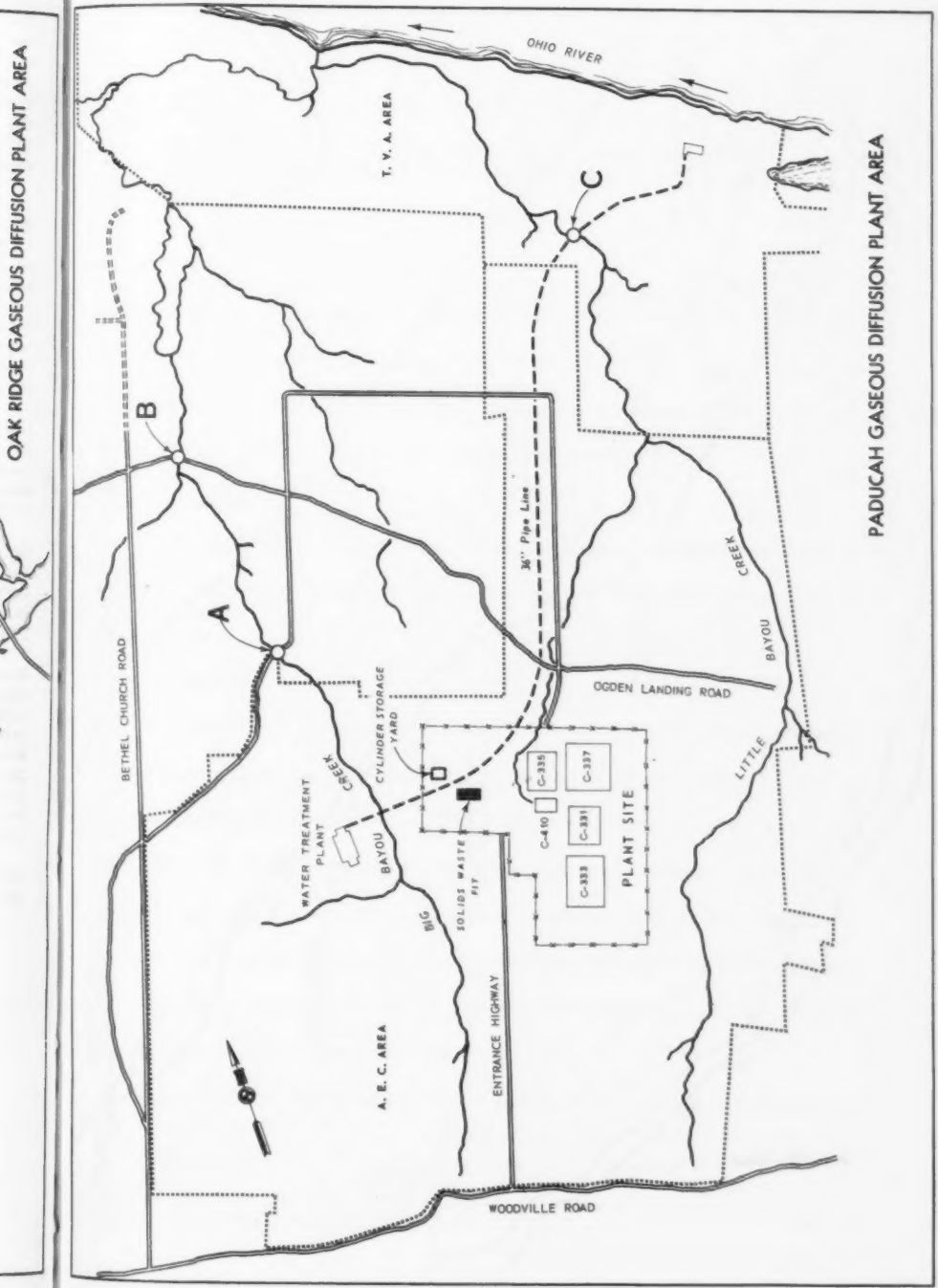
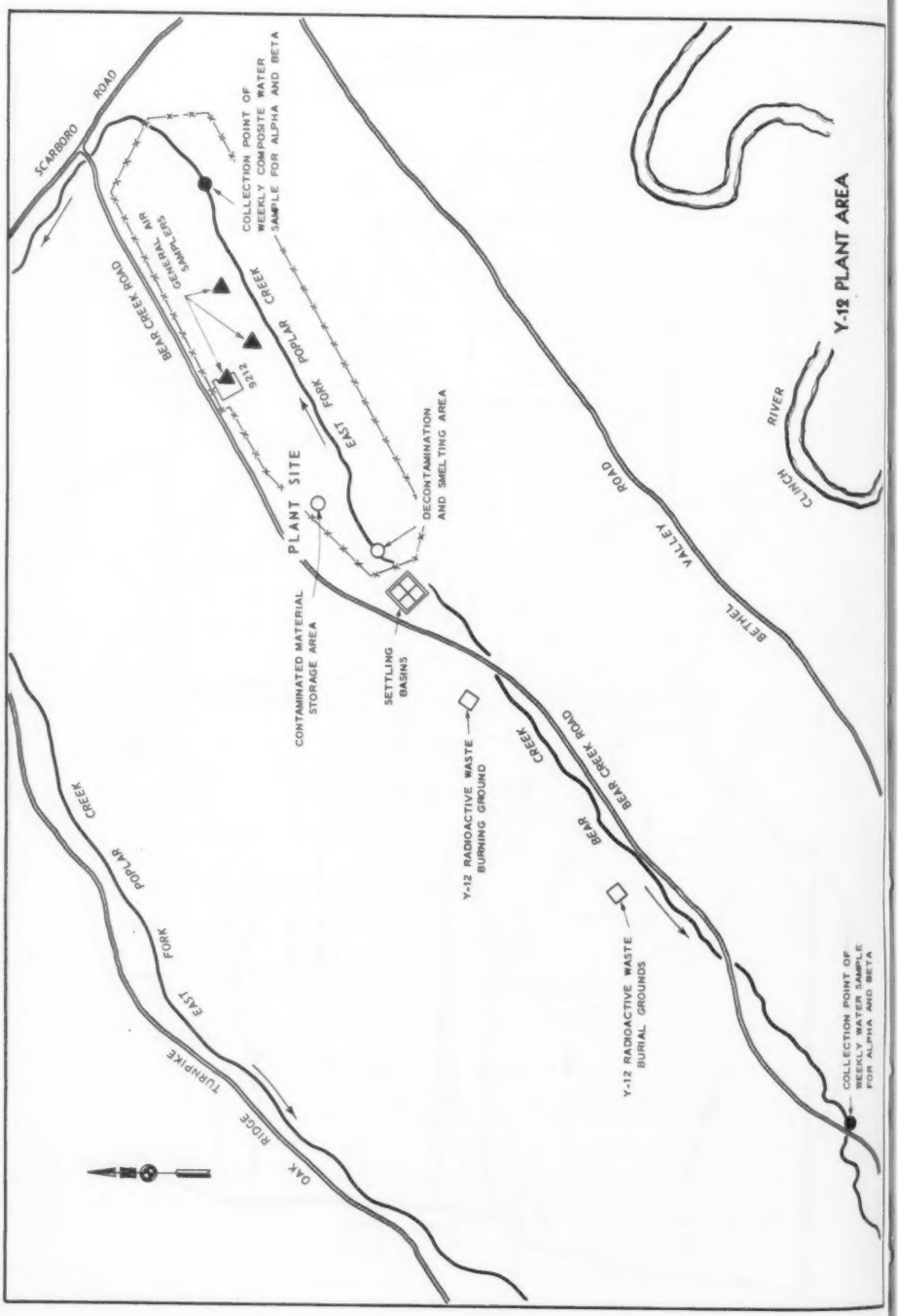

OAK RIDGE GASEOUS DIFFUSION PLANT AREA
BEAR CREEK VALLEY ROAD
OAK RIDGE TURNPIKE
BLAIR ROAD
DASHED LINE INDICATES OVERFLOW FROM K-1420 TO K-1407-B, SETTLING BASIN TO POPLAR CREEK.
AIR SAMPLING STATION FOR URANIUM
A
K-1407-B
K-1420
PLANT SITE
GENERAL AIR SAMPLERS
B
K-25
K-29
K-1131
K-27
K-33
K-31
ATMOSPHERIC VENT
POPLAR CREEK
CLINCH RIVER
POWERHOUSE
COLLECTION POINT FOR COMPOSITE WATER SAMPLE
CLINCH RIVER



OAK RIDGE GASEOUS DIFFUSION PLANT AREA


SCARBORO ROAD
COLLECTION POINT OF WEEKLY COMPOSITE WATER SAMPLE FOR ALPHA AND BETA
BEAR CREEK ROAD
SAMPLING GENERATORS AIR
POPLAR CREEK
EAST FORK
PLANT SITE
DECONTAMINATION AND SMELTING AREA
CONTAMINATED MATERIAL STORAGE AREA
SETTLING BASINS
Y-12 RADIOACTIVE WASTE BURNING GROUND
BEAR CREEK ROAD
BEAR CREEK
Y-12 RADIOACTIVE WASTE BURIAL GROUNDS
COLLECTION POINT OF WEEKLY WATER SAMPLE FOR ALPHA AND BETA
OAK RIDGE TURNPIKE
POPLAR CREEK
EAST FORK
BETHEL VALLEY ROAD
CLINCH RIVER
Y-12 PLANT AREA

Part I of Two Companion Papers

under the general heading --

"TREATMENT PLANT FOR REMOVAL OF RADIOACTIVE CONTAMINANTS FROM PROCESS WASTE WATER"

Part I "Engineering Design Features"
Mark C. Culbreath, m. ASCE, Principal Engineer, Burns and McDonnell Engineering Company, Kansas City, Missouri.

So far as is known this plant, located at Oak Ridge National Laboratory, is the first full scale unit of this type adapted for treatment of low level radioactive wastes, particularly for the removal of strontium.

Design Criteria

It is required to reduce the radiation activity to a level that will not exceed the acceptable standards for discharge into White Oak Creek and the Clinch River.

1. Treatment Requirements:

The treatment plant conforms very closely to that of a standard water softening plant, having the usual water softening equipment and using the usual water softening chemicals. The plant was designed with the requirement that it would be possible to reduce the calcium carbonate content to 30 to 35 ppm with a maximum of 50 ppm. Provision is made for feeding hydrated lime, soda ash, calcium chloride, clay and tri-sodium phosphate.

The plant influent will have a wide fluctuation in the pH value with a possible low of 2 and a possible high of 10.

The plant was designed with one treatment line capable of treating

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* Paper presented at the annual meeting of the American Society of Civil Engineers, October 13-17, 1958, New York, New York. Published in Journal of the Sanitary Engineering Division, Proceedings of the American Society of Civil Engineers, Vol. 85, 1959.

Y-12 PLANT AREA
COLLECTION POINT OF WEEKLY WATER SAMPLE FOR ALPHA AND BETA

500,000 gallons per day and so constructed that an additional treatment line of 500,000 gallons per day could be added at a later date

The detention time, at nominal rate, in the three units of the treatment line are as follows:-

Flash Mixer Basin	85 seconds
Coagulation Basin	31 minutes
Settling Basin	129 minutes

The overflow rate for the settling basin would be 695 gallons per square foot per day.

2. Plant Features:

Figure 1 Oak Ridge National Laboratory, shows the general area of the plant with the equalization basin in the foreground, the control building and the treatment plant which are the three main features we will be discussing in this paper.

The settling basin in the middle left is one of several original holding basins. It is still being used for wastes that do not require treatment. All process waste water containing low level radioactive waste from the various laboratory processes is either discharged to this settling basin or to the equalization basin for treatment in the plant. White Oak Creek lies just beyond the settling basin and the treatment plant.

The treatment plant may or may not be operated continuously, depending upon the amount of wastes that are being produced that require treatment. If the amount of these wastes are insufficient to require continuous operation of the plant, the wastes are stored in the equalization basin until such time

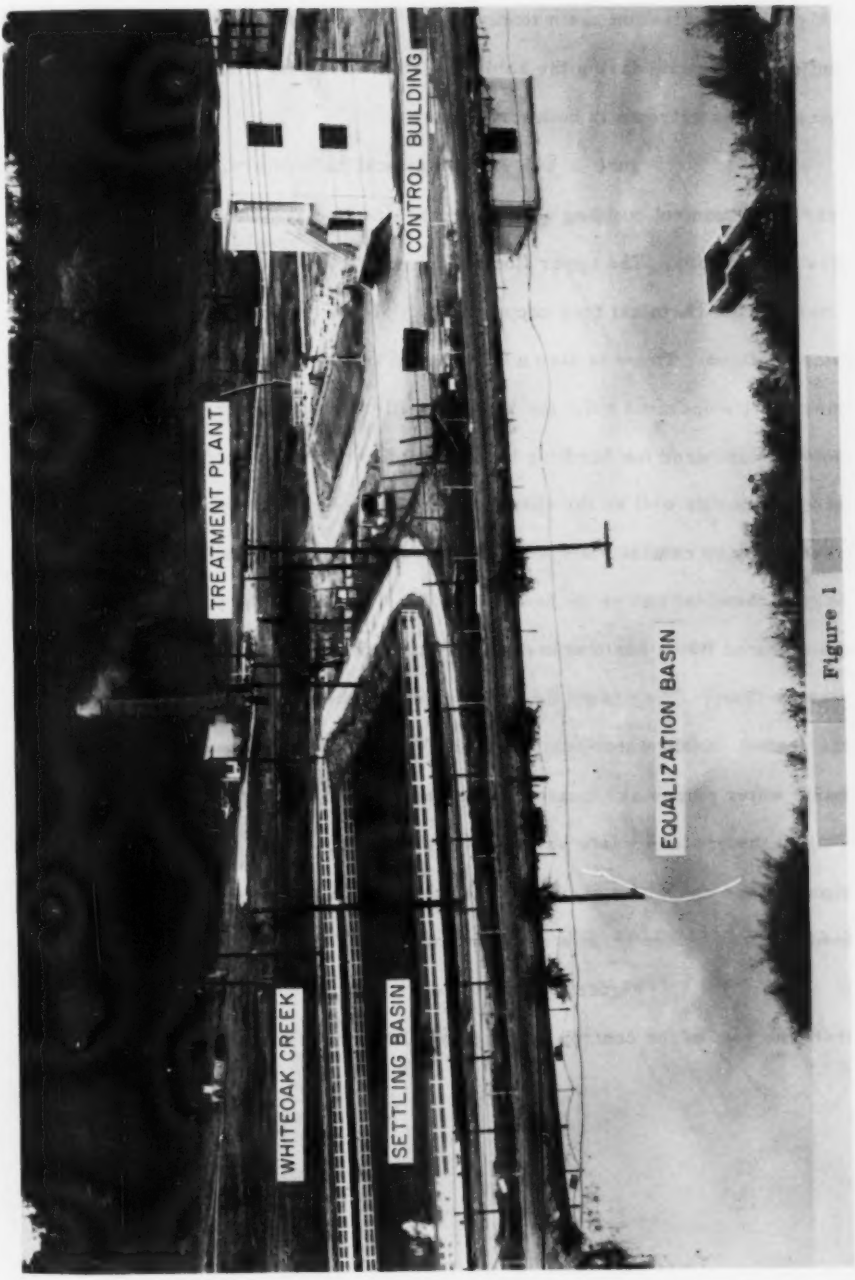
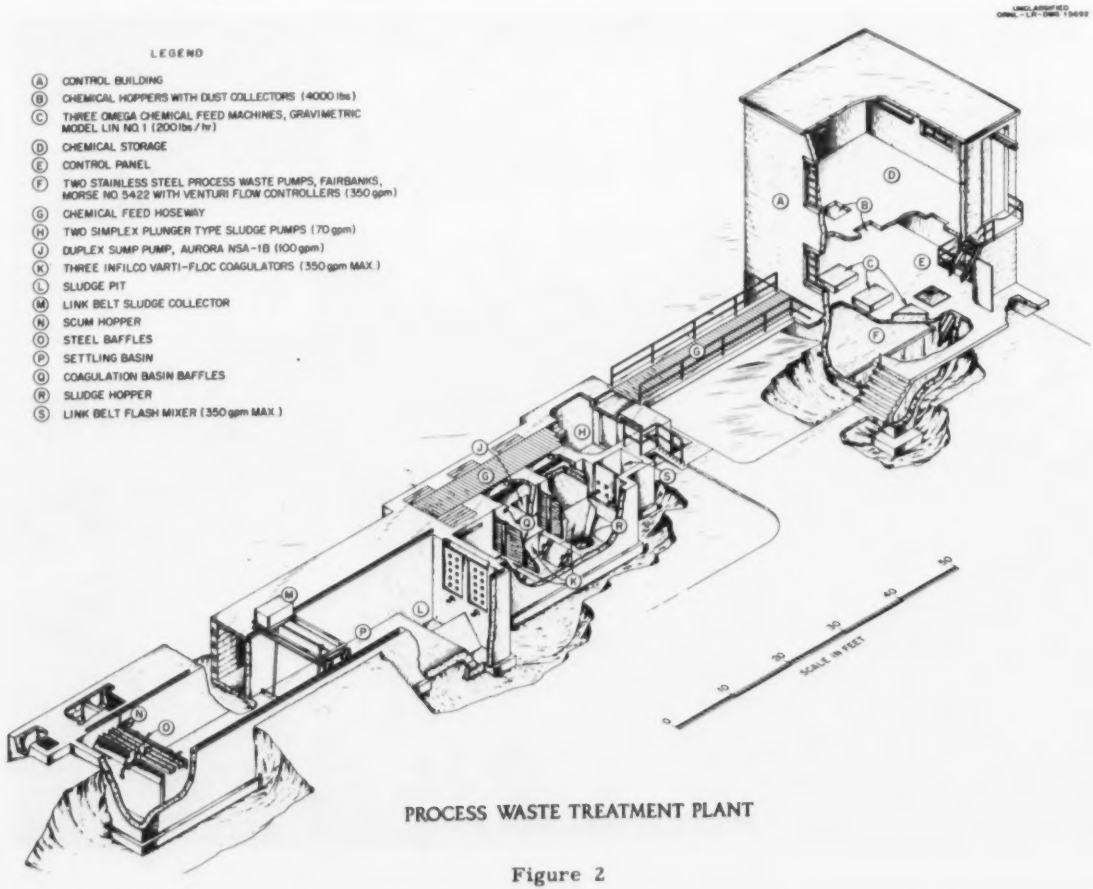

TREATMENT PLANT
CONTROL BUILDING
WHITEOAK CREEK
SETTLING BASIN
EQUALIZATION BASIN

Figure 1

as it is deemed advisable to treat them. The equalization basin has a capacity of approximately 875,000 gallons. Since the pH fluctuates considerably, the equalization basin tends to give the wastes a chance to become more uniform in quality during the holding period. That is, the pH will tend to become less extreme in both directions.

Figure 2 Oak Ridge National Laboratory, shows a cut-away of the control building and treatment line No. 1. The control building has three floors. The upper floor is for storage of bagged chemical and for charging the chemical feed hoppers which protrude above the chemical storage floor. There is also a "U" shaped trolley beam which carries an electrically operated hoist for handling pallets of bagged chemical. This hoist is also used for handling heavy equipment through hatchways in the ground floor as well as the chemical storage floor. The upper floor is reached by an outside stair from the loading dock. Trucks loaded with bagged chemical can be backed up to the loading dock, the pallets maneuvered into a position under the hook with a dolly and hoisted to the storage floor. The ground floor accomodates the chemical feed machines, the control panel, water heater, lavatory, sink and cabinets. The process waste water pumps are located in the basement of the control building as well as the process waste water piping and the steam and process water piping. The operators can reach the treatment plant by way of the chemical feed hoseway bridge or by a short walkway.

Figure 3 Oak Ridge National Laboratory, was taken from the roof of the control building and shows very clearly the treatment


LEGEND
A CONTROL BUILDING
B CHEMICAL HOPPERS WITH DUST COLLECTORS (4000 lbs)
C THREE OMEGA CHEMICAL FEED MACHINES, GRAVIMETRIC MODEL LIN NO.1 (200 lbs / hr)
D CHEMICAL STORAGE
E CONTROL PANEL
F TWO STAINLESS STEEL PROCESS WASTE PUMPS, FAIRBANKS, MORSE NO. 5422 WITH VENTURI FLOW CONTROLLERS (350 gpm)
G CHEMICAL FEED HOSEWAY
H TWO SIMPLEX PLUNGER TYPE SLUDGE PUMPS (70 gpm)
J DUPLEX SUMP PUMP, AURORA NSA-1B (100 gpm)
K THREE INFILCO VARTI-FLOC COAGULATORS (350 gpm MAX.)
L SLUDGE PIT
M LINK BELT SLUDGE COLLECTOR
N SCUM HOPPER
O STEEL BAFFLES
P SETTLING BASIN
Q COAGULATION BASIN BAFFLES
R SLUDGE HOPPER
S LINK BELT FLASH MIXER (350 gpm MAX.)
0
10
20
30
40
50
SCALE IN FEET

PROCESS WASTE TREATMENT PLANT

Figure 2

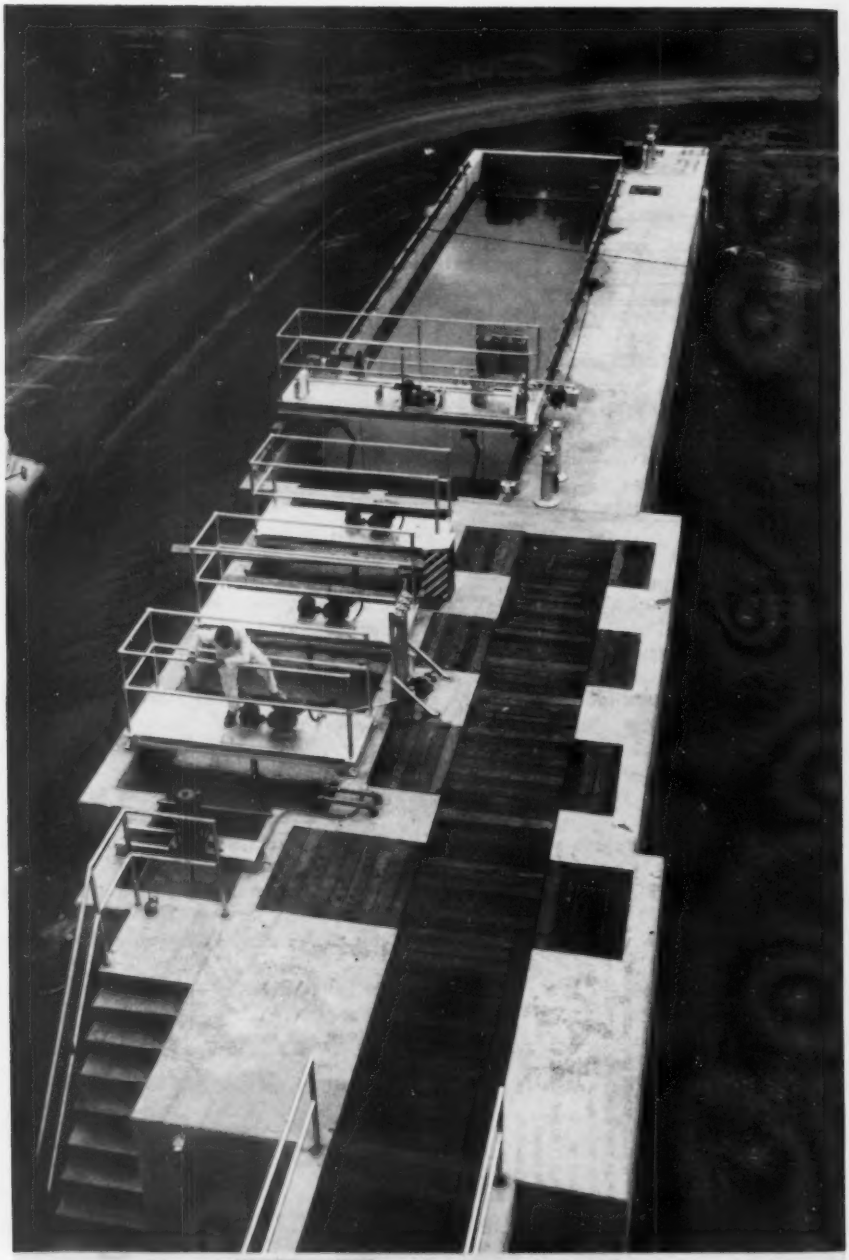


Figure 3

plant and the equipment being used.

In the future another treatment line will be added along the right hand side of the structure as viewed from this point. This, then, will result in one treatment line on each side of the hoseway and pipe gallery. This will give the plant a symmetrical shape.

In the center foreground is the chemical feed hoseway bridge between the control building and the treatment plant. The hoseway is covered with chemically treated wooden planks which may be removed as needed to service the chemical hose. There is a live steam line, uninsulated, which will give enough heat to prevent the hose lines from freezing during the mild winters of this area. The rectangular structure at the end of the hoseway bridge is the sludge pump room and directly below the pump room is located the sludge sump which has a capacity of 10,500 gallons. The sludge sump, being so deep in the ground, is well isolated by the ground and concrete.

Referring again to drawing number 15692, Oak Ridge National Laboratory, the first element in the treatment line is the flash mixer basin, followed by a three compartment coagulation basin which is 8' wide by 24' long and 9'-6" deep including 1'-4" of freeboard. The compartments are separated by wooden baffles and concrete fillets directly below the baffles since the baffles do not go all the way to the bottom.

These fillets are also provided along the side walls, thus giving the bottom of each compartment a sort of a hopper shape which, because of the rolling action created by the coagulators, are kept reasonably clean of sludge accumulations.

It is important to keep the basins as free as possible of sludge accumulations to prevent a build-up of radioactivity within them.

At the effluent end of the coagulation basin flat transite baffles are provided with circular holes in them, designed to smooth out and distrubute the flow into the upper end of the settling basin. At the junction of the coagulation basin and the settling basin, two sludge pits are provided which receive the sludge that settles out in the settling basin. It is also possible to flush out the coagulation compartments through slots provided in the fillets just below the baffles and openings in the wall between the coagulation basin and the settling basin. The settling basin is 60' long by 12' wide and 9'-6" deep, including 1'-4" of freeboard. Near the effluent end of the settling basin a scum hopper is provided for removal of scum which may possibly accumulate. It is not expected that there will be very much scum accumulation. At the effluent end of the settling basin a baffle is provided which is followed immediately by launders equipped with serrated weirs. Discharge from the launders is directly into the overflow chamber.

The above is a general view of the plant. The following portion will give more detail. Laboratory wastes originate in many different places. It was necessary, therefore, to collect all of those wastes--bringing them to a common point in so far as that could be done, so that they could be monitored to determine the amount of radioactivity they may be carrying. By use of pump stations and a common mixing box, it was possible to collect all of these wastes at one point except for one waste line. This waste line is pumped directly to the equalizing basin, since its activity is

too high for the settling basin. As the waste leaves the mixing box it passes through a monitoring station which measures the activity. Should the activity rise above a predetermined amount, the valves in the diversion box just beyond would be actuated automatically as to close the valve to the settling basin and open the valve to the equalizing basin. When the activity dropped below this predetermined value the reverse action would occur, closing the valve to the equalizing basin and opening the valve to the settling basin. From the diversion box the flow is either to the settling basin or to the equalizing basin splitter box. This splitter box will make it possible at a later date, when a second equalization basin is built, to divert the flow to either or both equalizing basins.

The equalization basin, as discussed previously, is primarily used for the smoothing out of the fluctuations in the flow of the process waste water. Also, it makes possible the intermittent operation of the plant. It also has a tendency to provide a more uniform quality to the treatment plant.

The process waste water pumps take suction from the equalization basin through an 8-inch pipe. The pipe originates in the basin suction sump, which is equipped with a stainless steel frame covered with a stainless steel wire mesh.

The control of the rate through the plant is set on the control board on the operating floor of the control building. It sets the rate of flow through the plant by controlling the butterfly on the downstream side of the Venturi meter, both of which are located in the pump discharge line. The flow goes directly to the flash mixer basin, and from there through a 1"

thick transite baffle which is provided with six 4" circular openings to smooth out the flow to the coagulation basin. From there it passes to the coagulation basin which is separated by wooden baffles into three compartments. As mentioned previously, part of this separation is accomplished by concrete fillets just below the baffles. These fillets are 2' high. The baffle between the first and second compartment is set 9" above the fillet and extends to the surface of the water, while the baffle between compartment 2 and 3 is set 1'-6" above the fillet and extends to the surface of the water. It is possible to apply chemicals at the flash mixer basin and at each of the coagulation basins as desired.

There are two baffles in the wall between the coagulation basin and the settling basin of 1" flat transite. Each of these baffles has ten 4" diameter openings in it to insure uniform flow into the settling basin. The settling basin, as was noted previously, is 12' wide and provided with sludge pits adjacent to the coagulation end. There are two sludge pits, either of which can be blown off separately to the sludge sump or returned, in any desired proportion, to the pump suction header and recycled through the treatment plant. The effluent is removed by way of launders equipped with serrated weirs at the effluent end of the basin. The flow from this point is directly into the overflow chamber, and from there to the effluent chamber, or returned for recycling to the pumps in the control chamber. Any desired percentage of recycling can be made.

3. Special Problems:

Provision has been made for the addition of a second equalization basin to the west of the present one, which when added, will give more flexibility to the plant as well as add to the equalization capacity.

Provision also has been made for adding a second treatment line of 500,000 gallons per day capacity just to the right side of the pipe gallery by merely knocking out a concrete block wall and adding the necessary basins. The present sludge pumps and pipe gallery will serve both.

Because of the very low pH which may at times occur, the pipelines from the equalization basin to the flash mixer basin are rubber lined. The first 48 feet of the 8-inch suction line from the suction sump in the equalization basin is rubber covered. This protection was added because of the possible deep penetration of the corrosive liquid into the earth embankment through which the pipe passes. The pumps, Ventrui meters, and the valves are of stainless steel. The valves are of the plug type and have rubber faced plugs.

The floor under the process waste water pumps is covered with stainless steel sheets which also extend up the wall a short distance. This provision makes it possible to maintain the utmost cleanliness. Corroded and dirty equipment permits radioactive build-up. Any drippings or wastes occurring during overhaul of pumps or other equipment, are taken through drains in the floor to the sump pumps in the pipe gallery adjacent to the coagulation basins. These wastes are pumped directly to the coagulation basin and processed. Likewise, the floor in the sludge pump room is

covered with stainless steel sheets, including the concrete foundations of the pumps, so that any wastes can be flushed to the sump pump and processed through the plant just as other wastes are handled.

Steam was piped to the control building and to the pipe gallery in the treatment plant for heating and for use in cleaning.

The basin equipment, including flash mixers, coagulators, and the sludge collector, were specified to receive at least two coats of polyethylene thermo-plastic coating after fabrication. This was done so that the equipment could be removed from its foundation by a crane and hosed down easily, while still being suspended above the basin, to prevent too much contamination adhering to the surface.

The sludge sump, as mentioned above, has a capacity of 10,500 gallons--which is several times that of the tank truck--which is used to remove the sludge from the site to an isolated waste area. A sampling cock is provided on the sludge pump discharge line for taking samples to determine the moisture content in the sludge. The sludge in the sludge sump may be made uniform, after periods of rest, by pumping the sludge up and returning it directly to the sump again. The sludge is pumped from the sump through an underground pipeline to a tank truck loading arm, similar to that used for oil services. The truck used is an oil type tank truck with a capacity of approximately 3,500 gallons. The operator can stop and start the pumps from a point outside of the sludge pump room, thus making it unnecessary for him to be in the room through which the radioactive wastes are pumped.

It will be noted by referring to the flow diagram (Fig. 4) that the layout will provide for the following operations:

a. May pump from the equalization basin to the treatment plant.

b. Pump from the equalization basin to the sewer.

c. Return any desired percentage of effluent for recycling.

d. Draw sludge from the settling basin sludge pits to the sludge pump.

e. Pump sludge from the sludge sump to the sludge tank truck.

f. Pump sludge from the sludge sump to the flash mixer.

g. Pump sludge from the sump back to it.

h. Pump sludge from the settling basin sludge pits directly to the sludge tank truck or to the flash mixer.

i. Blow off sludge from either settling basin sludge pit to the sludge sump.

j. Draw sludge from the settling basin sludge pits through the effluent return line to the process waste water pumps for recycling.

k. Pump from the treatment basins back to the equalization basin .

After the second equalization basin is constructed, and treatment line No. 2 added, the following additional operations will be possible in addition to the above:

l. Operate treatment line No. 1 and treatment line No. 2 in series. or parallel.

m. Recycle both treatment lines in parallel to any desirable percentage.

n. Pump from one equalization basin to the other.

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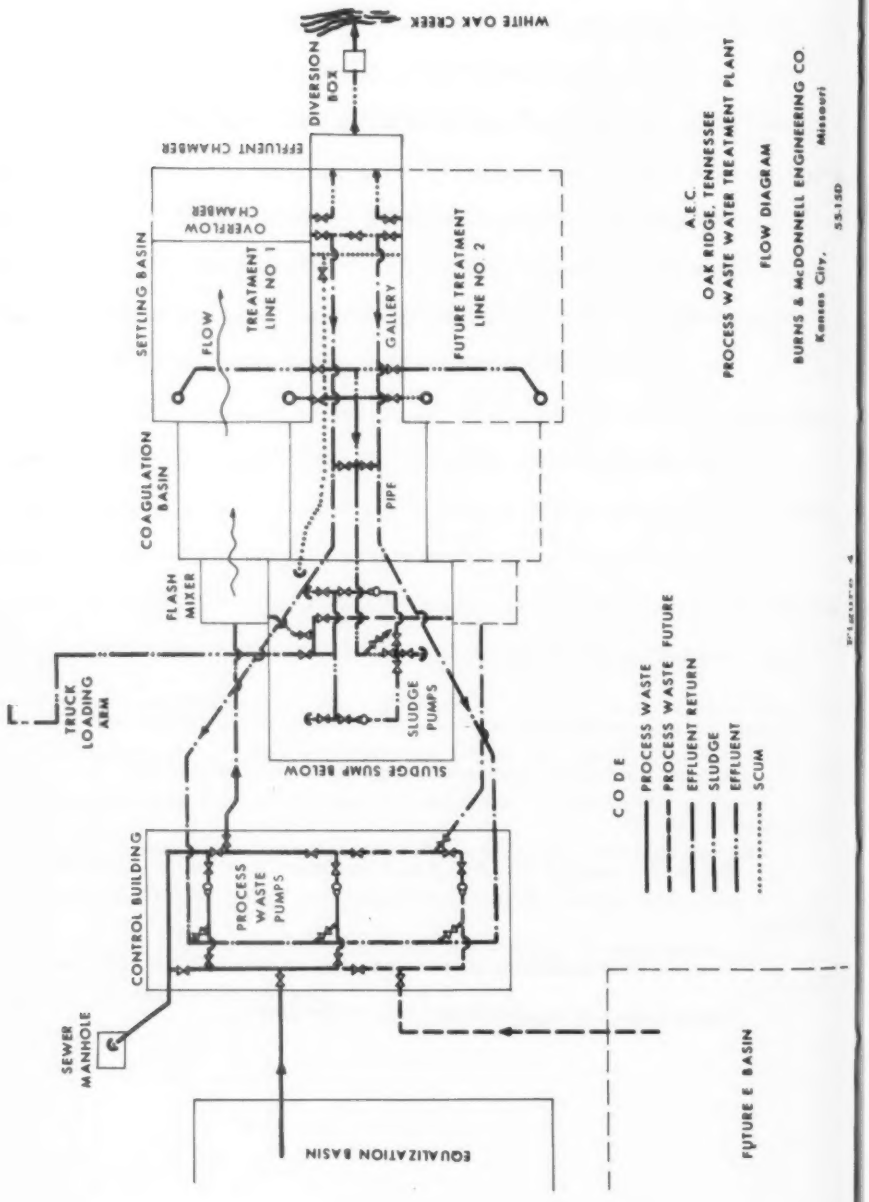

WHITE OAK CREEK
DIVERSION BOX
EFFLUENT CHAMBER
OVERFLOW CHAMBER
SETTLING BASIN
TREATMENT LINE NO 1
FLOW
GALLERY
FUTURE TREATMENT LINE NO 2
COAGULATION BASIN
PIPE
FLASH MIXER
TRUCK LOADING ARM
SLUDGE PUMPS
SLUDGE SUMP BELOW
CONTROL BUILDING
PROCESS WASTE PUMPS
SEWER MANHOLE
EQUALIZATION BASIN
FUTURE E BASIN
CODE
PROCESS WASTE
PROCESS WASTE FUTURE
EFFLUENT RETURN
SLUDGE
EFFLUENT
SCUM
A.E.C.
OAK RIDGE, TENNESSEE
PROCESS WASTE WATER TREATMENT PLANT
FLOW DIAGRAM
BURNS & McDONNELL ENGINEERING CO.
Kansas City, Missouri
55-15D

In summary it should be pointed out that there are three main features of the plant that make it different:

First - The provision for a wide range of flexibility in operation which gives it the character of a laboratory.

Second - The provisions for maintaining cleanliness by the use of:

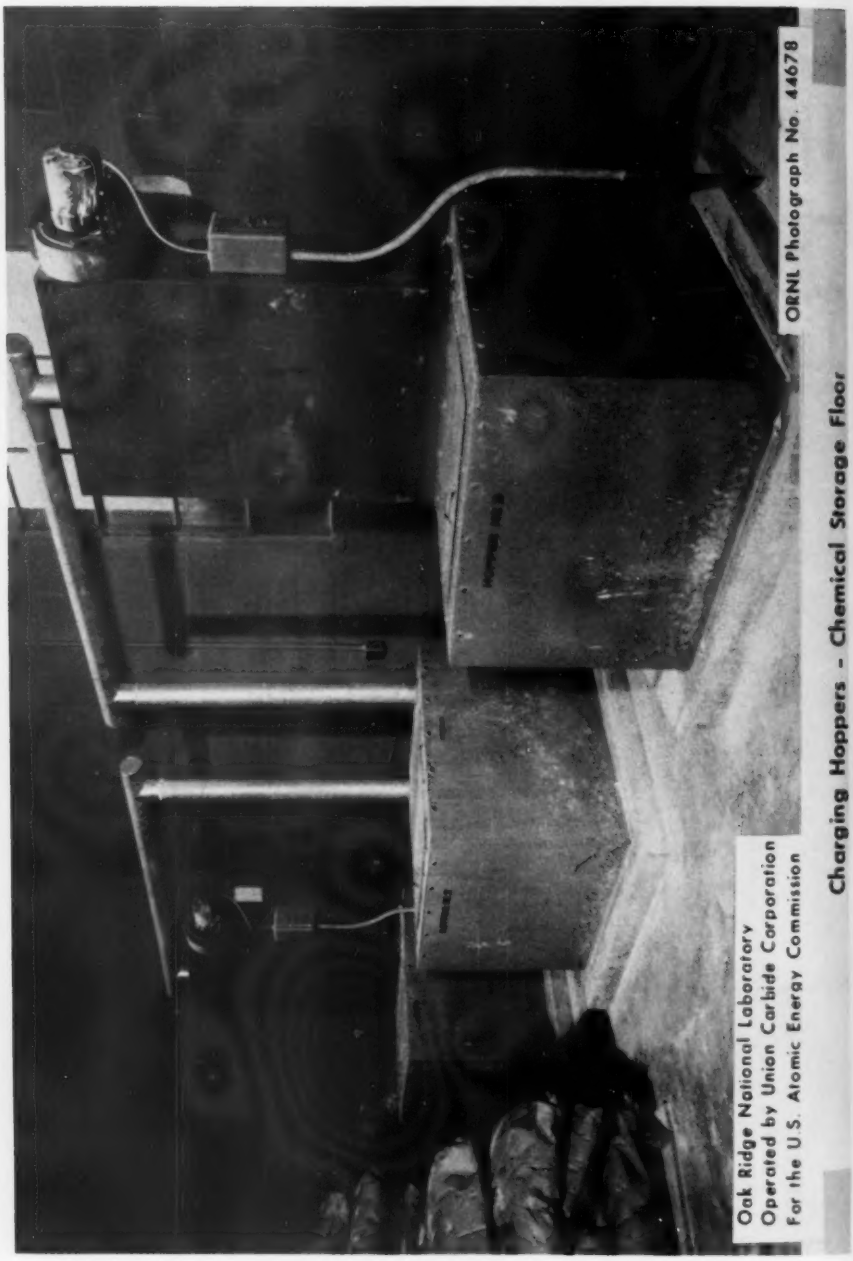
a. Stainless steel floors.

b. Collecting drips and spills for reprocessing.

c. Hosing down of equipment as it is being removed for maintenance, and

d. By so constructing the basins as to prevent excessive accumulations of sludge.

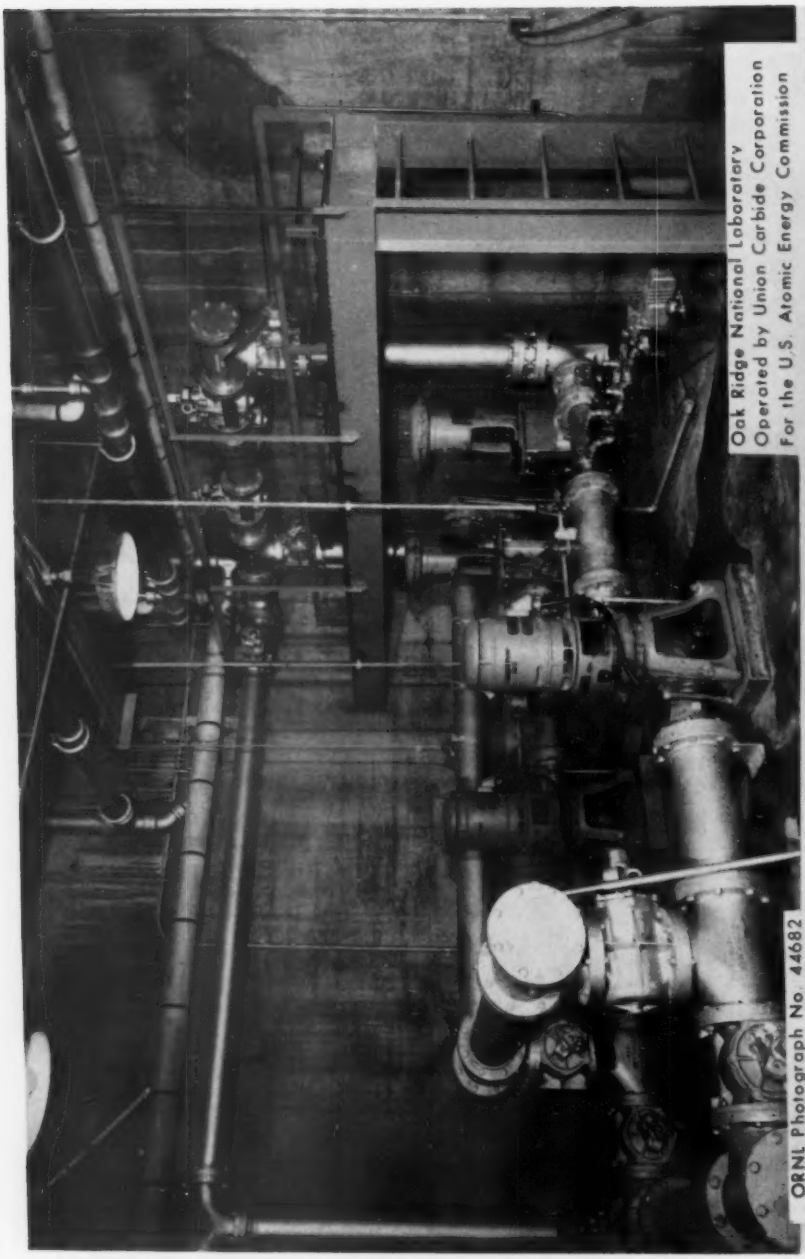
Third - By providing for maintenance of basin equipment from above, thus avoiding the necessity of draining the water from the basins which is a very good shield against radiation.


Oak Ridge National Laboratory
Operated by Union Carbide Corporation
For the U.S. Atomic Energy Commission
Charging Hoppers - Chemical Storage Floor
ORNL Photograph No. 44678

Oak Ridge National Laboratory
Operated by Union Carbide Corporation
For the U.S. Atomic Energy Commission
ORNL Photograph No. 44677

Chemical Feeders - Operating Floor

Charging Hoppers - Chemical Storage Floor


Oak Ridge National Laboratory
Operated by Union Carbide Corporation
For the U.S. Atomic Energy Commission
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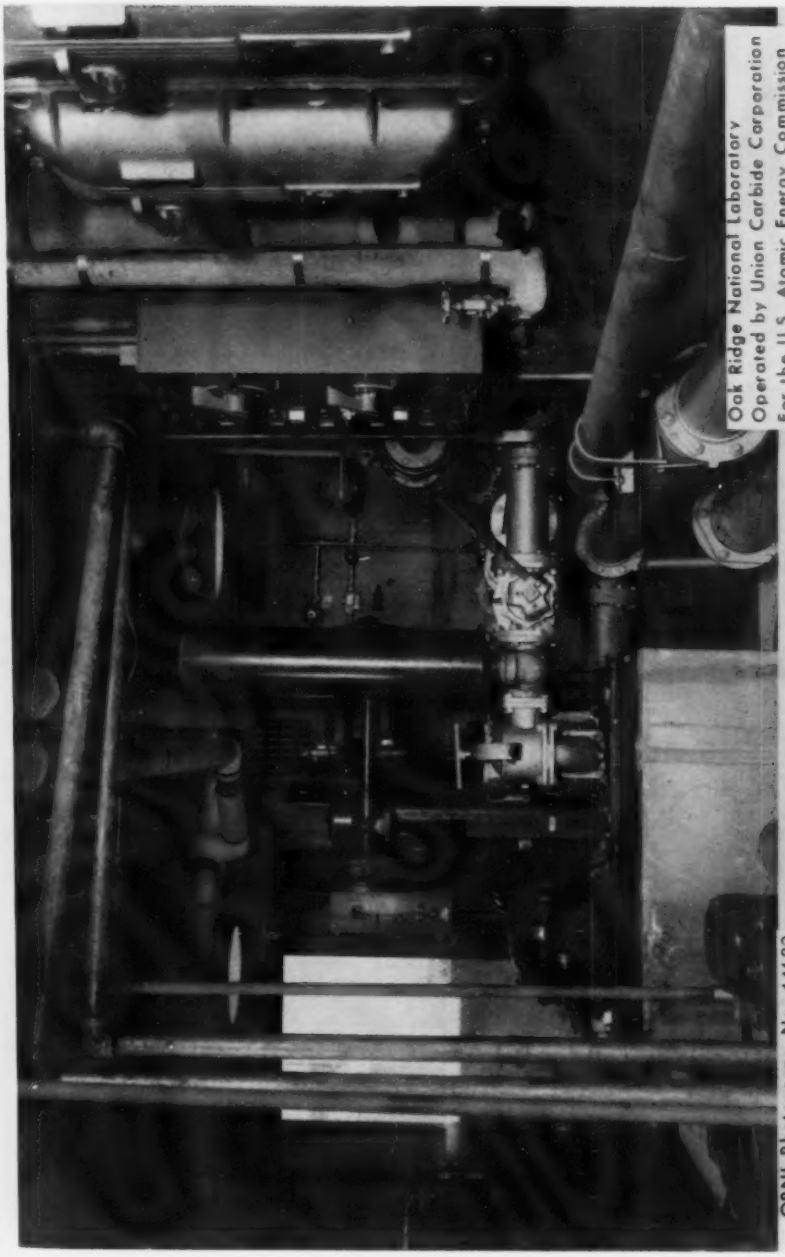
Process Waste Pumps and Venturi- Basement Floor


Oak Ridge National Laboratory
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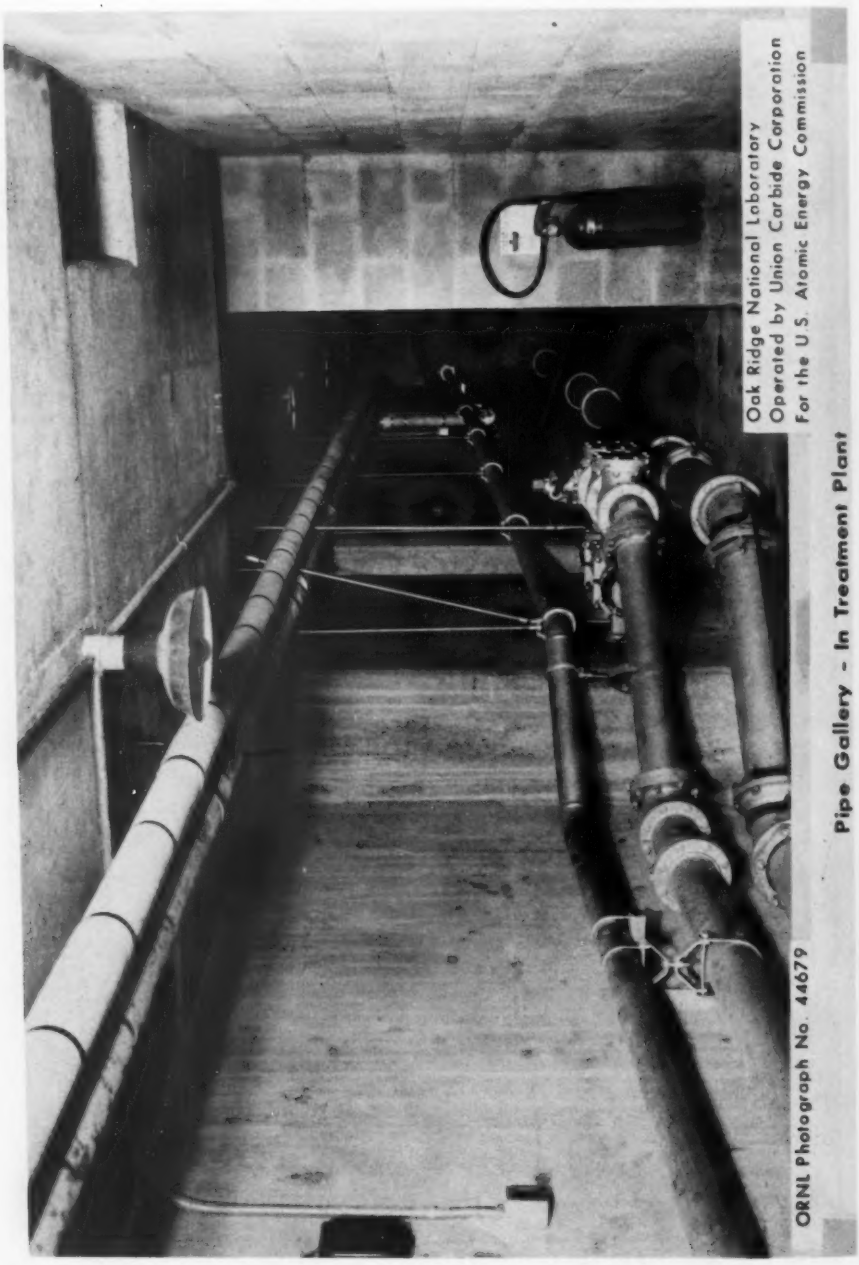
ORNL Photograph No. 44680

Sludge Pumps - In Treatment Plant

Process Waste Pumps and Venturi- Basement Floor


Oak Ridge National Laboratory
Operated by Union Carbide Corporation
For the U.S. Atomic Energy Commission
ORNL Photograph No. 44679
Pipe Gallery - In Treatment Plant

TREATMENT PLANT FOR REMOVAL OF RADIOACTIVE CONTAMINANTS FROM PROCESS WASTE WATER*

Part II: Evaluation of Performance

By K. E. Cowser and Roy J. Morton**

The safe and economical disposal of process waste water presents a problem at all atomic energy installations. These wastes are usually low in radioactivity but large in volume. However, the total quantity of radioactive constituents is often too great to be diluted sufficiently in the environment; uncontrolled release is not desirable. When treatment of the waste is required: (1) the process should concentrate the major portion of the radioactive contaminants to facilitate safe and permanent disposal; (2) the process should be sufficiently flexible for adaptation to changes in flow and chemical characteristics of the waste; (3) the process should provide a reserve capacity for more intensive treatment in case of accidental spills or emergency releases; and (4) the cost per unit volume of waste treated should be low.

Types of Radioactive Liquid Wastes

At Oak Ridge National Laboratory (ORNL) there are three liquid waste streams that contain radioactive materials. These are:

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*Paper presented at the annual meeting of the American Society of Civil Engineers, October 13-17, 1958, New York, New York. Published in Journal of the Sanitary Engineering Division, Proceedings of the American Society of Civil Engineers, Vol. 85, 1959.

**Oak Ridge National Laboratory, operated by Union Carbide Corporation for the U. S. Atomic Energy Commission, Oak Ridge, Tennessee. Included work by H. H. Abee, R. L. Newton, and H. J. Wyrick.

Pipe Gallery - In Treatment Plant

(1) Process wastes (contaminated waste water) in volumes of 200-300 million gallons per year which contain between 200 and 500 curies of radioactivity per year. A discussion of these wastes and their treatment is the main purpose of this paper.

(2) Intermediate-level wastes (referred to as chemical wastes at ORNL) which average 2 to 3 million gallons per year with beta activities of about 10^5 to 10^6 counts per minute per ml (c/m/ml).(a) These wastes are held in underground tanks for neutralization, followed by disposal to waste pits which utilize the sorptive capacity of an indigenous shale formation.[1, 2]

(3) Metal wastes containing recoverable heavy metals and radioactivity of intermediate levels (as in 2 above). These are held in underground tanks for neutralization and metal recovery. The liquids containing the fission products are disposed of in the same way as the intermediate-level chemical wastes.

THE PROCESS WASTE SYSTEM

These three types of liquid waste are collected in separate systems. Radioactivity monitoring points are provided, and the different wastes are directed from their sources to the appropriate system for handling and disposal.

Process wastes come from equipment cooling water, floor drains, decontamination pad drains, storage canals, laboratory sinks, and miscellaneous low activity operations. A network of 6-inch to 30-inch diameter

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(a) All results are referred to second-shelf counting using an end-window Geiger counter.

vitrified clay pipe is used to collect and convey the wastes by gravity flow through a central monitoring point to the settling basin or to the equalization basin. The system is divided into three main sections which, in turn, are subdivided by means of monitoring stations. These stations are concrete manholes in which a V-notch weir, float recorder, and a Sigma pump for the collection of samples are located. A flow diagram of the waste system is shown in Figure 1.

The volumes and radioactivity levels of the process waste during the years 1950 to 1957 are summarized in Table I. The total volume varies from about 165 to 298 million gallons per year, or an average of 0.45 to 0.81 million gallons per day (mgd). The fourth column in Table I, headed "Retention Pond," refers to radioactivity contributed by the drainage from monitoring pads underneath the underground waste storage tanks. The total gross beta activity discharged from the process waste system and the retention pond to White Oak Creek varies from 172 to 498 curies per year.

The effluent from the settling basin is measured and composite samples are collected by a continuous proportional sampler at the weir box[(b)] near the point of discharge to the Creek. Radiochemical analyses of these samples for five years (1953-1957) are summarized in Table I. Most of the radioactivity in process waste is due to trivalent rare earths (including cerium), cesium, and strontium; lesser amounts are contributed by ruthenium, zirconium, iodine, niobium, barium, and cobalt.

The location and surface water drainage system of the Laboratory and its environs is shown in Figure 2. The wastes discharged to White

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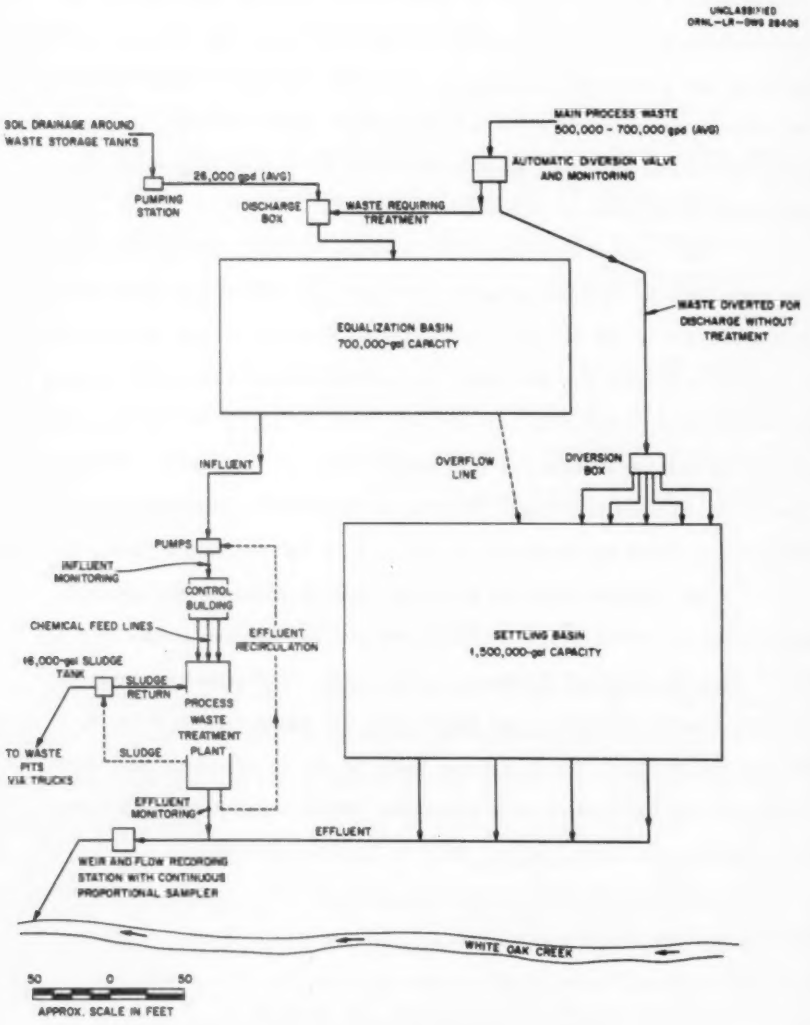
(b) Weir box is equipped with: (1) Standard 90° V-notch weir; (2) Standard USGS installation of water level recorder; and (3) Trebler sampler supplied by Lakeside Engineering Corporation, Chicago, Illinois.



Flow Diagram of Process Waste System.

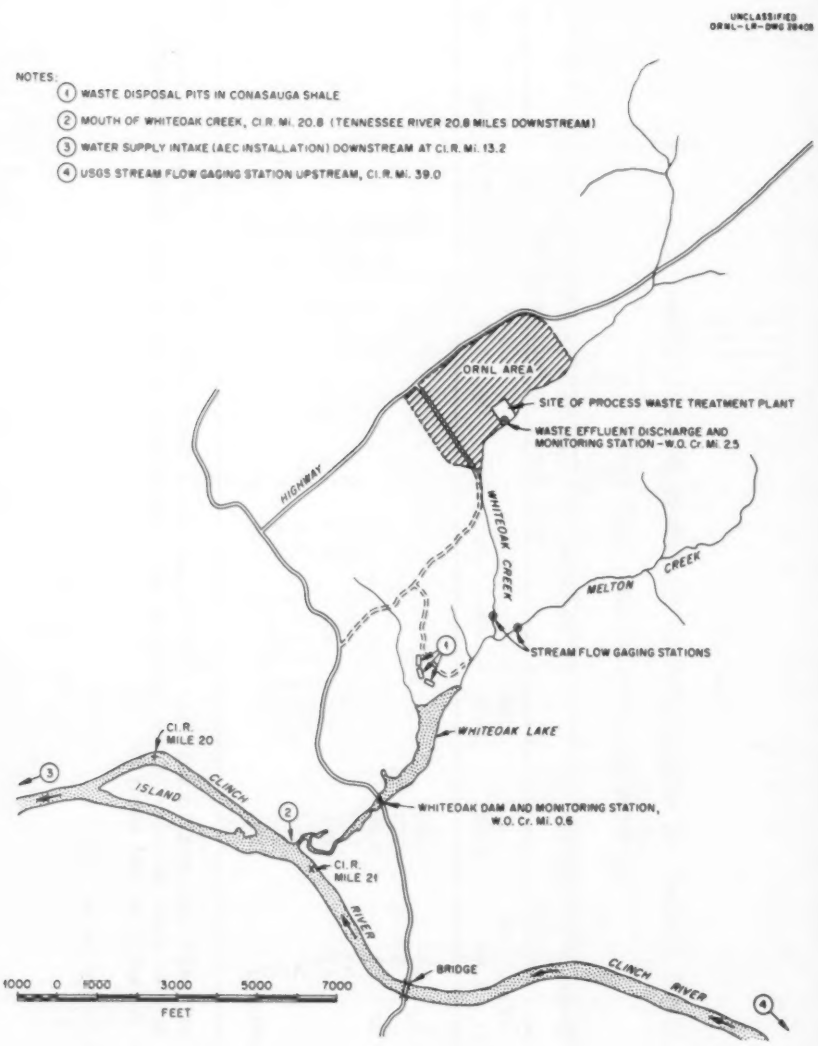
TABLE I

YEARLY VOLUMES AND RADIOACTIVITY OF DISCHARGES OF LIQUID WASTES TO WHITE OAK CREEK AT ORNL, 1950-1957[a]

Year	Total Volume 10^6 gals	Gross Beta Activity Curies			Per Cent of Gross Beta Activity Identified with Specific Radionuclides								
		Settling Basin	Retention Pond	Total	TRE[b] (-Ce)	Ce	Ru	Zr	Cs	I	Sr	Nb	Ba	Co
1950	226.4	172	15	187										
1951	297.6	169	3	172										
1952	268.2	411	87	498										
1953	239.4	289	140	429	53.1	2.6	0.8	2.6	8.5	0.2	27.6	1.0	0.8	---
1954	164.3	237	17	254	34.6	19.1	0.5	1.0	20.3	0.5	23.3	0.4	0.3	---
1955	210.6	213	54	267	30.3	14.7	3.1	0.6	31.6	0.2	18.7	0.6	0.2	---
1956	260.7	253	20	273	24.4	12.3	2.0	0.5	42.1	0.1	15.1	1.0	0.2	2.2
1957	272.3	(combined)		189	25.8	4.4	1.0	10.2	36.5	0.0	18.0	0.7	0.0	1.8

a. Volume of settling basin effluent measured in weir box with liquid level float recorder. For determinations of radioactivity continuous proportional samples of the effluent are composited for daily gross beta measurements and monthly radiochemical separations and analysis.

b. Trivalent rare earths exclusive of cerium.


UNCLASSIFIED
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NOTES:
1 WASTE DISPOSAL PITS IN CONASAUGA SHALE
2 MOUTH OF WHITEOAK CREEK, CI.R. MI. 20.8 (TENNESSEE RIVER 20.8 MILES DOWNSTREAM)
3 WATER SUPPLY INTAKE (AEC INSTALLATION) DOWNSTREAM AT CI.R. MI. 13.2
4 USGS STREAM FLOW GAGING STATION UPSTREAM, CI.R. MI. 39.0
ORNL AREA
SITE OF PROCESS WASTE TREATMENT PLANT
WASTE EFFLUENT DISCHARGE AND MONITORING STATION - W.O. Cr. Mi. 2.5
HIGHWAY
WHITEOAK CREEK
MELTON
CREEK
STREAM FLOW GAGING STATIONS
WHITEOAK LAKE
CI.R. MILE 20
ISLAND
CLINCH
WHITEOAK DAM AND MONITORING STATION, W.O. Cr. Mi. 0.6
CI.R. MILE 21
RIVER
BRIDGE
CLINCH RIVER
1000 0 1000 3000 5000 7000
FEET

Location Sketch Map
ORNL Area Surface Drainage

Oak Creek flow approximately three miles through the Creek channel, the basin of White Oak Lake, White Oak Dam, and the embayment above the mouth of the Creek. The average annual rate of flow of the Creek into the Clinch River, obtained from data at gaging stations on White Oak Creek and Melton Creek, is approximately 5 cubic feet per second (cfs). Occasionally, heavy floods cause rates of flow in excess of 1000 cfs for several hours. The Creek discharges into the Clinch River at Cl. R. Mi. 20.8.(c) The mean annual flow in the Clinch River is approximately 5000 cfs which provides a dilution factor of about 1000 for the water and wastes discharged from White Oak Creek. Conditions of disposal into the Clinch River have been described in greater detail elsewhere.[3] It is the policy to comply with the recommendations of the International Commission on Radiological Protection regarding radioactive contamination in drinking water for nonoccupational and large population use.[4]

BASES OF DESIGN OF TREATMENT PLANT

The experience of the water works profession in the purification of domestic water supplies has demonstrated that many nonradioactive contaminants can be removed from water by various water treatment procedures at moderate cost. Since 1948, studies of the decontamination of low-level contaminated water have been conducted at ORNL and elsewhere. These studies have included the evaluation of conventional and nonconventional water treatment processes for removal of radioactive materials from water and the development of methods for increasing their efficiency.[5, 6, 7, 8, 9, 10] The

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(c) Cl. R. Mi. = Clinch River Mile - distance in miles numbered upstream from mouth of River.

results of these studies suggested that modified water treatment procedures would be suitable for the treatment of ORNL process waste.

A number of water treatment processes were investigated to determine their removal efficiencies for Sr^{90}, Cs^{137}, and the rare earths, from tap water. The characteristic efficiencies of six treatment processes are shown in Table II. A prime consideration in the evaluation of each process was its ability to remove up to 90% of the radiostrontium. Sand filtration and chemical coagulation with aluminum and iron salts were unsatisfactory for the removal of strontium and cesium. Lime-soda softening, phosphate coagulation, and ion exchange were all found to be capable of removing more than 90% of the strontium and the rare earths.

The laboratory studies were extended to include actual process wastes at ORNL. Up to 90% removal of gross radioactivity could be obtained with excess lime-soda softening or phosphate coagulation when clay was added for the removal of cesium. Although phosphate coagulation was promising, efficient removals of strontium required accurate control of pH (to approximately 11.5) and of the ratio of phosphate (Na_3PO_4) and lime ($Ca(OH)_2$) dosages.

Increased removals of radiostrontium by the lime-soda softening process were found to be related to: (1) reduced concentration of calcium in the effluent; (2) excess dosages of lime and soda ash (above stochiometric requirements); and (3) successive precipitations of calcium in the presence of previously formed calcite crystals.

Pilot plant tests with process wastes were performed using three types of upflow coagulation and sedimentation units having rated capacities of 6 and 35 gallons per minute (gpm) and an ion exchange assembly utilizing

TABLE II

REMOVAL OF RADIONUCLIDES FROM WATER BY CONVENTIONAL WATER TREATMENT PROCESSES

Radioisotope	Process Waste Stream Composition (per cent of gross beta 1954-1956)	Per Cent Removal by Treatment Process					
		Chemical[1] Coagulation	Chemical Coagulation Plus 100 ppm Clay	Sand Filtration	Lime-Soda Softening (150 ppm excess)	Phosphate Coagulation (240 ppm dose)	Ion Exchange
Sr	19.6	3	0-51	4	97.3	97.8	99.1-99.97
Ce	15.2	91	85-96	---	----	99.9	----
Trivalent Rare Earths (including Y)	30.4	91	----	87	90.0	----	75.0-98.7
Cs	29.9	0.5	35-65	50	Not Effective	----	99.8
Ru	1.9	77	----	---	----	----	----
	97.0						

(1) Coagulant includes alum, ferrous sulfate or ferric chloride, lime, soda-ash or sodium hydroxide, and sodium silicate.

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a cationic resin. The results showed that the upflow units employing the lime-soda softening process removed greater than 50% of the gross activity, but that there was too much variation in removal efficiency due to changes in the chemical composition of the waste stream. Also upflow units were not well adapted for repeated precipitation. The use of ion exchange was discouraging because of cost, the apparent need for pretreatment, the relatively short runs before regeneration was required, the variable chemical composition of the waste, and the operational difficulties experienced at the Laboratory with other ion exchange units.

The over-all results of the laboratory and pilot plant studies indicated that the process waste treatment plant should be a horizontal-flow, lime-soda water softening process with provisions for alternative use of phosphate coagulation for strontium removal and the addition of clay to increase the removal of cesium. To reduce the concentration of radioactivity from accidental or emergency releases, recirculation of the effluent for repeated treatment would be required.

The criteria of design and the major features of the treatment plant are reported in Part I, "Engineering Design Features," by M. C. Culbreath, Principal Engineer, Burns and McDonnell Engineering Company.

OPERATIONAL EXPERIENCE

The process waste treatment plant was completed in August 1957. With the new treatment plant available, process wastes can be held and treated as intensively as necessary, provided the level of activity is within the capability of the plant to achieve the required decontamination.

A new treatment plant usually requires a break-in period in which to reach optimum operation. During this period the operating personnel are trained, the mechanical components of the plant are adjusted; the limitations of the process are established; the proper combinations of chemical dosages are determined; and controls are established. This treatment plant is now in the break-in period and will remain so until these problems have been resolved. This paper describes the operating experience and removal efficiencies obtained during the first eleven months of operation.

Wastes requiring treatment are collected in the equalization basin of 700,000-gallon operating capacity (see Figure 1). This basin serves to minimize fluctuations in composition of the plant influent and to supply waste for treatment. The wastes are bypassed automatically at the central monitoring station when the level of radioactivity is so low that treatment is not required. The by-passed wastes flow through the settling basin before discharge to the Creek. The effluent from the treatment plant joins the effluent from the settling basin and the combined discharge is monitored near the point of release to White Oak Creek.

From September 1957 through January 1958 the plant was operated 24 hours a day and 5 days a week. A 7-day operating week was initiated February 1958. Wastes were generally bypassed to the settling basin during the late evening and early morning hours when the major processes which produce radioactive wastes were not in operation. The automatic diversion system was placed into operation May 12, 1958. Since then, all wastes generated from 8:00 a.m. to 8:00 p.m. have been collected in the equalization basin; from

8:00 p.m. to 8:00 a.m., wastes are selectively bypassed, according to their level of radioactivity, by the automatic diversion equipment.

The components of the continuous water monitor and appurtenant diversion equipment have been described by Gillespie.[11] As shown in Figure 3, the detector consists of five halogen-filled Geiger-Müller (GM) tubes enclosed in celluloid centrifuge tubes and suspended within a one-gallon polyethylene container. A sample at a rate of one gallon per minute is pumped through the polyethylene container and back to the waste stream. Radioactivity detected by the GM tubes in the sample is continuously recorded, and a switch arrangement on the recorder actuates a motor-operated diversion valve. The sensitivity of the monitor to Sr^{90} - Y^{90} in equilibrium is 6.8×10^{-6} microcuries per milliliter. Based on MPC values for drinking water, Sr^{90} with its daughter Y^{90} is the most critical fission product in the waste.

The plant cannot treat all of the process waste. To optimize the removal of fission products from the total volume of waste, it is necessary to treat that portion of the waste having the greatest concentration of radioactivity. The minimum concentration of radioactivity required before waste is diverted to the settling basin must be determined by experience. A trial and error method is necessary because of the variable rate of waste release, the fluctuating concentrations of radioisotopes in the waste, the gross radioactivity determinations made by the water monitor, and the limited capacity of the equalization basin and treatment plant.

The rate of flow through the plant is kept constant at 300 gallons per minute. About 6 gallons per minute of water is added to this flow by each chemical feeder. During the first five months of plant operation 43 million gallons of process waste amounting to 34% of the total volume was

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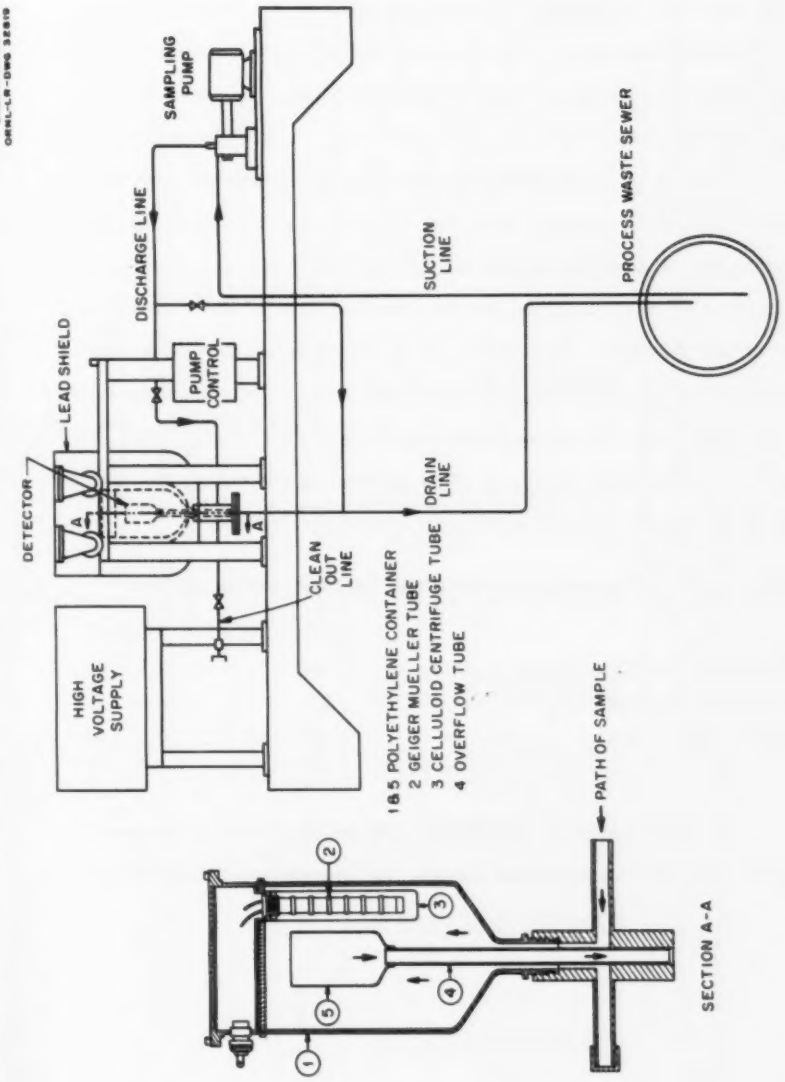

FIGURE 3.—Process Waste Water Monitor Assembly

treated. In the next six months 76.6 million gallons of waste was treated, accounting for 67% of the total.

Waste Composition and Chemical Requirements

Routine chemical analyses, including tests for pH, alkalinity, calcium, and total hardness, are performed daily on the raw and treated waste. Continuous samples of the influent and effluent are collected and composited for weekly and monthly radiochemical analyses. Samples are analyzed for gross beta activity, Sr^{89}, Sr^{90}, Cs^{137}, total rare earths (TRE), Ru^{106}, and Co^{60}. Specific radionuclides are separated from the composite samples by chemical techniques.[12] Absorption studies are used to distinguish between the amounts of Sr^{89} and Sr^{90}, and an empirical correction curve is used to adjust for the counting loss of Sr^{90} by carrier solids on the planchet. The observed count rates of the separated radionuclides of cesium, ruthenium, and cobalt are converted to disintegration rates by comparison with standardized solutions of Cs^{137}-Ba^{137m}, Ru^{106}-Rh^{106}, and Co^{60}. An efficiency factor of 10 is assumed in converting the TRE from c/m/ml to disintegrations per minute per milliliter (d/m/ml).

The chemical composition of the plant influent was as follows:

	Average	Range
Calcium hardness (ppm as $CaCO_3$) -------	86	33 to 135
Total hardness (ppm as $CaCO_3$) ---------	127	59 to 180
Alkalinity (ppm as $CaCO_3$) ------------	94	40 to 326
pH ----------	7.3	2.5 to 11.5

The concentrations of radionuclides in the waste are given in Table III. For the eleven months listed, the average concentrations of

TABLE III

CONCENTRATION OF RADIONUCLIDES IN TREATMENT PLANT INFLUENT AND EFFLUENT[a]
(Monthly Composite Samples)

Date	Influent							Effluent						
	Gross Beta Activity	Sr^{89}	Sr^{90}	TRE	Cs^{137}	Co^{60}	Ru^{106}	Gross Beta Activity	Sr^{89}	Sr^{90}	TRE	Cs^{137}	Co^{60}	Ru^{106}
Sept. 1957	110	41	280	220	240	30	80	36	5.2	34	23	210	13	74
Oct. 1957	64	13	200	200	180	23	33	21	3.7	55	48	150	18	34
Nov. 1957	160	16	210	180	150	13	70	20	3.2	43	36	120	3.8	12
Dec. 1957	70	12	190	210	130	17	33	14	1.9	32	27	80	3.0	11
Jan. 1958	77	16	310	250	180	17	40	36	4.1	80	66	---	3.9	14
Feb. 1958	46	16	140	170	150	14	21	20	5.0	45	57	90	6.2	12
Mar. 1958	77	17	270	300	170	12	23	21	2.6	42	37	110	3.6	8.0
Apr. 1958	79	13	260	280	170	15	15	24	2.1	42	40	100	3.9	4.0
May 1958	94	15	340	340	320	25	24	39	3.4	73	67	320	9.6	7.3
June 1958	89	14	260	260	300	17	17	39	2.3	43	38	320	5.6	13
July 1958	140	21	520	400	480	20	10	63	3.2	81	55	500	5.8	7.1

(a) Gross beta activity in units of c/m/ml referred to second-shelf counting using an end-window Geiger counter. Sr^{89}, Sr^{90}, TRE, Cs^{137}, Co^{60}, and Ru^{106} in units of d/m/ml.

radionuclides in the influent were: 18 d/m/ml of Sr^{89}, 270 d/m/ml of Sr^{90}, 260 d/m/ml of TRE, 220 d/m/ml of Cs^{137}, 18 d/m/ml of Co^{60}, and 33 d/m/ml of Ru^{106}. The compositing of monthly samples has an averaging effect on the concentration of activity as seen by the gross beta activity in weekly samples which has ranged from 21 c/m/ml to 260 c/m/ml.

During this period only lime and soda ash have been used continuously in the treatment process. The studies of McCauley[8] indicated the need for reducing calcium to low levels for efficient strontium removal, while those of Hoyt[10] suggested that an excess dosage of lime and soda ash was instrumental in removing greater quantities of strontium. Therefore, various combinations of lime and soda ash were tried. Theoretically, the amount of material added is sufficient to satisfy the stoichiometric requirements of the raw waste and also provide a pre-established excess of lime and soda ash in the system. The amount of lime is based on the consumption of lime by carbon dioxide, carbonate hardness, magnesium hardness, and the excess dosages required. Sufficient soda ash is added to combine with the noncarbonate hardness and excess lime, and to maintain an excess of carbonates in the waste. The calculations are based on the equations describing the reactions in the lime-soda process given by Black.[13]

Efficiency of Plant Operation

From the analyses of composite samples of the plant influent and effluent (Table III), the efficiency of the treatment process is determined. The percentage removal of the important radionuclides for monthly periods is given in Table IV. Strontium removals ranged from 66% to 87% and have been as high as 94% for individual weeks. The total rare earths were

TABLE IV

REMOVAL OF RADIONUCLIDES BY TREATMENT

		Percentage Removals							Curies Removed				
		Monthly Composite Samples					Weekly Composite Samples		Monthly Composite Samples				
Date	Gross Beta Activity	Sr^{89+90}	TRE	Cs^{137}	Co^{60}	Ru^{106}	Sr^{89+90}	TRE	Sr^{89+90}	TRE	Cs^{137}	Co^{60}	Ru^{106}
Sept. 1957	67	87	89	9.5	51	7.6	88	89	3.7	2.6	0.62	0.02	0.19
Oct. 1957	68	73	77	19	23	0	73	73	2.6	2.7	0.84	0.12	0
Nov. 1957	87	80	80	21	72	83	75	82	2.4	1.9	0.23	0.12	0.74
Dec. 1957	79	84	87	40	83	67	83	85	2.4	2.6	0.74	0.21	0.31
Jan. 1958	53	74	74	----	76	65	71	75	3.6	2.7	----	0.19	0.38
Feb. 1958	57	66	67	42	56	42	65	65	2.3	2.4	1.4	0.17	0.18
Mar. 1958	73	85	88	32	69	66	84	85	4.7	5.1	1.1	0.16	0.30
Apr. 1958	70	84	86	40	74	74	84	87	5.3	5.5	1.6	0.25	0.26
May 1958	59	78	80	0	61	69	86	86	6.0	5.9	0	0.34	0.36
June 1958	57	84	86	0	67	24	90	89	5.2	5.2	0	0.25	0.09
July 1958	55	85	86	0	70	30	78	81	10	7.6	0	0.30	0.06
							Total -------		48	44	6.5	2.1	2.9

removed as efficiently as the strontium. The reduction of gross beta activity ranged from 53% to 87% and was dependent on the concentration of radionuclides in the waste and the type and amount of chemicals used. The increased reduction of gross activity during November 1957 was partially attributable to the efficient removal of Pa^{233} from the waste. This was indicated by the fact that the sample of influent for November contained 240 d/m/ml of Pa^{233}, while it was not detected in the sample of the plant effluent.

To test the reliability of the analytical techniques employed for strontium and total rare earth determinations, the differences in percentage removals calculated from weekly and monthly composite samples (see Table IV) were treated statistically. The differences in percentage removals as calculated were not different from zero, even at the 80% level of significance.

Since clay has not been added on a continuous basis cesium removals were low as was anticipated. During the week of December 9, 1957, 30 ppm of Conasauga shale(d) were added to the waste, and cesium removals increased from about 16% to 55%. The affinity of the shale for cesium was determined by Jacobs and Tamura[14] to be due to the illitic type of clay mineral that comprises a large fraction of the shale. Voids created by the nonexpanding lattice structure of illite will accommodate ions of the size of cesium ions. Guided by this knowledge, a commercial clay product of high illite composition was recently procured, and addition of this clay

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(d) Conasauga shale dried and ground - containing 40% illite, 25% montmorillonite, and 35% silica.

to the treatment process begun.(e)

Weekly composite samples were analyzed for calcium hardness and total hardness, as well as for strontium and the total rare earths. The percentage removals of strontium and the total rare earths were graphed as a function of the residual calcium remaining in the plant effluent are shown in Figure 4 and Figure 5. As calcium residuals increase, the percentage removal of strontium decreases, and the apparent relationship is expressed as a coefficient of correlation of -0.74; the 95% confidence limits are -0.57 and -0.86. Similarly, the coefficient of correlation between calcium residuals and the total rare earths is -0.66 with 95% confidence limits of -0.45 and -0.79. The real meaning of these relationships can be determined only after the mechanism of strontium and total rare earth removals has been established. Although the amount of precipitated magnesium is about 7% of the precipitated calcium, the carrying capacity of magnesium hydroxide for strontium and the total rare earths must be determined. There is no apparent relationship between the amount of calcium carbonate that is precipitated and the removal of strontium.

By using an excess of soda ash above that required by the non-carbonate hardness and excess lime, increased removals of calcium were obtained. This would be expected from the common-ion effect, since calcium, in the presence of excess carbonate, must be correspondingly reduced in order to maintain the ion product equal to the solubility constant of calcium carbonate. At the present time excess lime is not used but 200 ppm of excess

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(e) Grundite: produced by the Illinois Clay Products Company, Joliet, Illinois - containing 65 to 75% illite, 10 to 20% kaolinite, and 5 to 15% quartz.

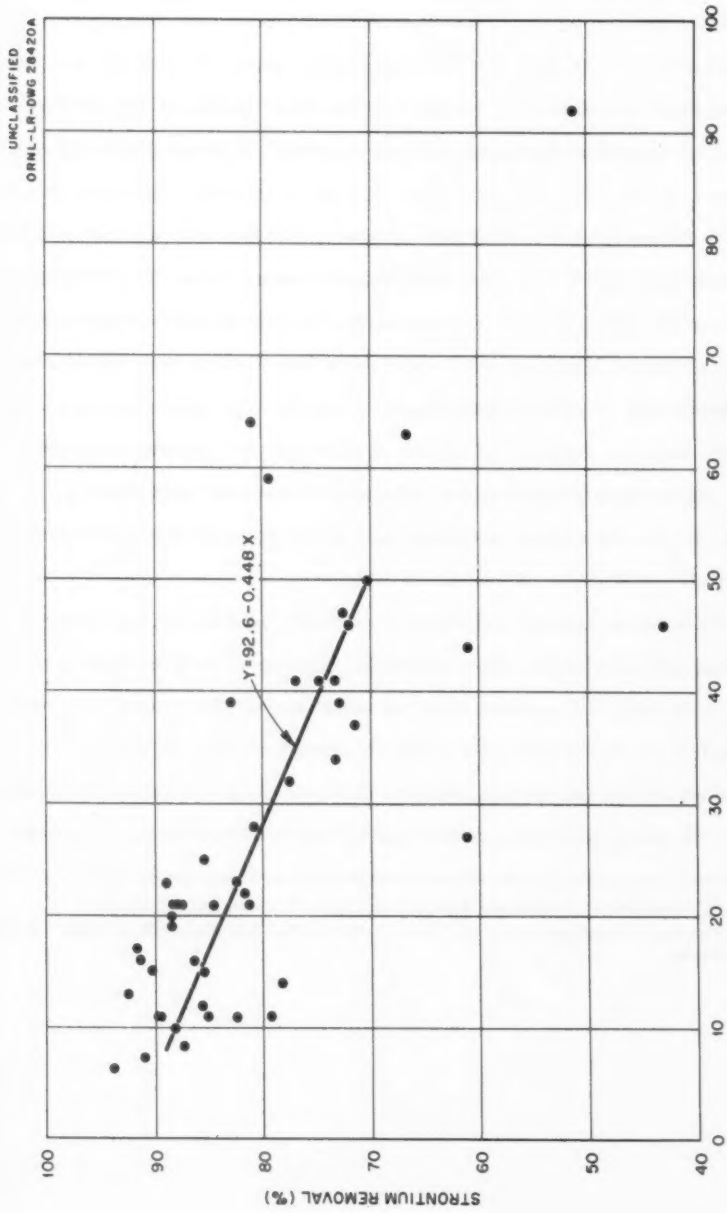

Fig. 4 Relation of Ca^{++} Residual to Strontium Removal.

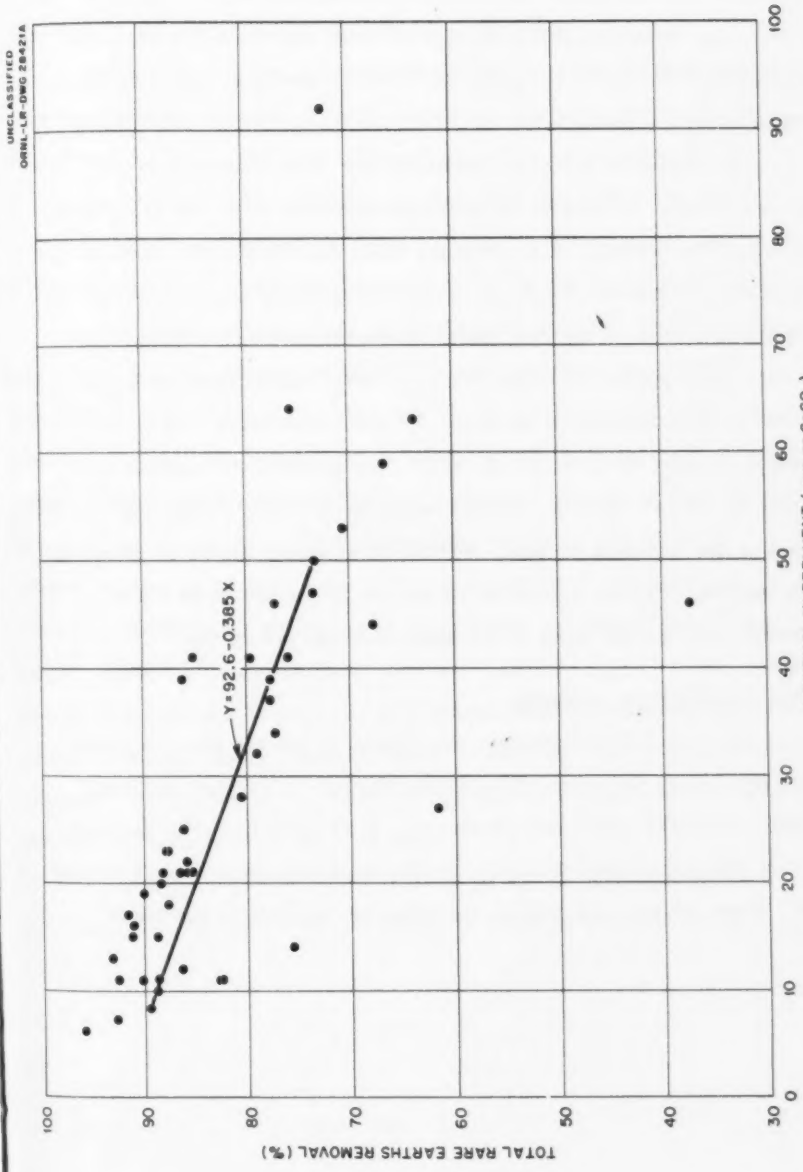

FIG. 5 Relation of Co^{++} Residual to Total Rare Earths Removal.

soda ash are being used.

A relationship exists between the per cent removal of strontium and the total rare earths and is expressed as a coefficient of correlation of 0.89. The 95% confidence limits for the correlation coefficient are 0.81 and 0.94. With percentage removals of total rare earths as ordinates and percentage removals of strontium as abscissa, the relationship is given by the equation $Y = 13.3 + 0.856\ X$.

The curies of individual radionuclides removed each month are shown in Table IV. A total of 48 curies of strontium and 44 curies of total rare earths were removed. Since February 1958, when continuous operation of the plant was started, about 67% of the process waste was treated and 67% of the strontium in the total waste stream was removed. Although this would imply 100% removal of strontium, it actually demonstrates that waste collected by the equalization basin had a higher concentration of strontium than waste bypassed to the settling basin. The radioactivity removed is contained in the calcium carbonate and magnesium hydroxide sludge that is produced in the treatment process. For the first eleven months of operation 387,000 gallons of sludge containing an average of 52% solids by volume were transferred by tank truck to the waste disposal pit system.[2]

Difficulties with Plant Operation

Low temperatures influence the removal of calcium from the waste. During cold weather the amount of calcium remaining in the plant effluent increases. With additional time of contact, up to fifty hours, it has been found that the calcium in solution or in fine colloidal suspension is reduced by 50%. Since the settling tank of the plant was designed to provide a

contact time of 2.15 hours, some means of increasing the rate of reaction that results in the precipitation and settling of calcium carbonate is necessary. Recycling previously settled sludge from the settling basin to the flash mixer in experimental amounts of 200 ppm was found to be effective in aiding the reduction of calcium residuals.

Complexing agents, such as versene, are used for decontamination purposes. Versene is known to chelate calcium, magnesium, and strontium, in preferential order, and most efficiently in alkaline solution.[15] In one instance the presence of versene was believed responsible for inefficient removal of calcium from the waste. About 550 ppm of sludge was recycled through the plant, but did not improve the removal of calcium.

For short periods during the early morning hours, the suspended solids in the plant effluent increase. In some instances the amount of suspended material leaving the plant was as high as 10% of the total solids that were formed. The percentages of magnesium hydroxide and calcium carbonate of the suspended material were not defined. The use of coagulants to aid in the flocculation of these suspended solids is one method for eliminating this difficulty. The proper selection of a coagulant aid depends on knowledge of the charge of the colloid. Although Clarke[16] and Larson[17] agree that magnesium hydroxide has a positive charge, Clarke reports that calcium carbonate bears a positive charge, while Larson reports that calcium carbonate bears a negative charge. Thus, the first step in selecting a coagulant aid is the determination of the type of colloid and its charge under the conditions in which it is formed.

During a period of three days the pH of the plant influent was less than 3.0. The low pH was the result of dropping the level in the

equalization basin to only one foot, thereby reducing the volume of waste that would act as a buffer for small releases of acid. Now a minimum depth of four feet of waste is maintained in the equalization basin at all times. Lime feed was inadvertently increased from 200 to 400 ppm, while the soda ash feed remained at 200 ppm for the first day of the period of low pH. The pH was increased to 11.5, but residual calcium was in excess of 150 ppm. By reversing the amount of lime and soda ash introduced, calcium residuals were reduced below 35 ppm.

Mechanical equipment at the plant has operated satisfactorily. By experience it has been determined that sludge draw-off from the collecting sump in the settling basin should be operated each morning for about three-fourths of a minute. The impeller of the flash mixer and the flash mixing basin are steam-cleaned once a week to prevent the build-up of sludge.

Plant Control

The results of these first eleven months of plant operation suggest that the amount of calcium in the effluent is indicative of the efficiency of removal of strontium and the total rare earths. Therefore, high softening efficiency appears desirable. Since a calcium determination can be made rapidly, it may serve as a simple test to control the treatment process. Laboratory studies show that an increase in the pH of the treated wastes results in a decrease of the residual calcium and an increase in the per cent removal of strontium. A pH in excess of 10.5 appears desirable, and this is in accord with the findings of McCauley[8] and Hoyt.[10] However, control of plant operation by pH determinations alone is inadequate. During periods when the temperature of the waste was less than 9° C and versene

was detected in the waste, the pH of the plant effluent averaged about 11.0; the average amount of calcium remaining in solution was greater than 57 ppm, and the removal of strontium was decreased.

The high degree of correlation between the removal of strontium and the total rare earths provides a basis for estimating removals of the rare earths with fewer analyses. The cause and effect relationship points to the removal of strontium and the rare earths by coprecipitation. However, the exact mechanism of removal remains to be determined. Nevertheless, the routine analysis of weekly composite samples for the rare earths has been discontinued; analysis for this group of isotopes will be continued in the monthly composite samples.

The plant was designed to allow the addition of various amounts and combinations of lime, soda ash, clay, returned sludge, and calcium chloride. It is desirable to determine the optimum dosage for each of these materials which will remove the greatest amount of radioactive strontium and cesium. Since a trial and error attempt to vary five factors on an operating scale is difficult to control, further laboratory studies are necessary.

To determine the importance of each of these five variables, separately and in combination, and the characteristics of their response surfaces, Gardiner[18] constructed a statistical experiment referred to as a Second Order Rotatable Design.[19, 20] Fewer experiments are required to attain the desired information. The design provides for testing the effect of each variable independently, in combination with each of the other variables, and to detect curvature in the response surface of each variable.

While the study has not been completed, preliminary results are available. The first series of jar tests using typical process waste were performed in two groups; the first group contained 23 beakers and the second group 10 beakers. The range of concentration of each variable was based on experience in plant operation and prior laboratory studies. The range of concentrations used were as follows:

lime ----------	stoichiometric requirement to 210 ppm excess
soda ash ----------	stoichiometric requirement to 300 ppm excess
sludge (calcium carbonate) --	0 to 2000 ppm
calcium chloride ----------	0 to 10 ppm in each of three additions
clay (Conasauga shale) ------	0 to 200 ppm

The results of these tests showed that the removal of strontium was influenced primarily by the addition of soda ash and the removal of cesium was influenced by the addition of clay. The effects of soda ash on strontium removal and clay on cesium removal are directly proportional and significant at the 5% level. Guided by the results of the first 33 tests, 10 additional tests were performed. Lime dosage was varied from the stoichiometric requirement to one-third of the requirement; soda ash was varied from the stoichiometric requirement to 375 ppm excess dosage; and clay type (Conasauga shale and Grundite), dosage, and particle size were also varied. Soda ash and lime (less than stoichiometric dosage) affected strontium removals, while clay type, dose, and size affected cesium removal. All of the effects were significant at the 5% level. As the lime dosage was increased to the stoichiometric requirement strontium removals were increased. Soda ash dosage greater than stoichiometric increased strontium removals. Grundite was 11% more effective on the average than Conasauga shale in the removal of

cesium. Also, the removal of cesium increased as dose increased and particle size decreased. Additional tests will be run employing combinations of lime, soda ash, and Grundite (particle size and dosage) that should lead most rapidly to optimum removals of strontium and cesium.

The laboratory study described above will furnish information to optimize the removal of strontium for the readily observable and controllable variables, lime and soda ash. However, the basic mechanism responsible for the removal of strontium in ORNL process waste must be determined before strontium removals can be maximized. The use of illitic type clay mineral for the removal of cesium exemplifies the improvement that can be made in a treatment process when the mechanism of removal is understood.

Information obtained from operation of the treatment plant and laboratory studies indicate that the removal of strontium by the precipitation of calcium carbonate is due to coprecipitation. Hahn[21] categorized four types of coprecipitation: (1) the formation of isomorphous mixed crystals where foreign ions are incorporated isomorphously in the crystal lattice; (2) surface adsorption by the precipitate after it has already been formed; (3) internal adsorption of impurities during formation of the crystals; and (4) the formation of anomalous mixed crystals.

An understanding of the coprecipitation phenomenon responsible for the removal of strontium may allow adjustment of plant operation to maximize removal of the radioisotope. For example, as pointed out by Bonner and Kahn,[22] if coprecipitation is by mixed crystal formation, rapid precipitation from a supersaturated solution followed by the immediate removal of the precipitate from its mother liquor is the most efficient process. If surface adsorption

is the primary mechanism of removal, allowing the precipitate to remain in contact with the solution will decrease the amount of strontium removed. Therefore, prompt removal of sludge from the settling basin is desirable. However, if internal adsorption is the responsible mechanism for removal, the fraction of tracer removed may increase as the precipitate remains in contact with the solution. Prompt removal of sludge may, therefore, be detrimental to maximum plant efficiency. Fundamental studies of the ORNL system are necessary before the mechanism of strontium removal can be determined.

Cost of Treatment

The unit cost for treatment of the process waste is difficult to determine, since it is influenced by a number of factors, some of which are variable or indefinite. With only limited experience in operation of this plant, absolute values could not be placed on certain factors, and these were not included in the calculations. It should be recognized that the costs summarized here are incomplete. They may be substantially greater when based on a longer period of operation and a more complete analysis of all operational requirements.

The total cost of chemicals used in treatment of the process waste for the first forty-seven weeks of operation is based on unit costs of lime and soda ash at $18.50 per ton and $46.40 per ton, respectively. The cost of operators' time, $300 per week, includes overhead and is based simply on a percentage of the total man hours worked by the operator. Utility costs average $15 per week. Maintenance costs, analytical costs, and amortization of the initial investment in the treatment plant are not included as a part

of the treatment costs. Thus, the total expenditure for the initial forty-seven weeks of plant operation was $21,500. During this period 118.7 million gallons of waste were treated. The cost per 1000 gallons of waste treated was $0.18.

CONCLUSION

The process waste treatment plant has demonstrated that the lime-soda softening process is an effective and economical method for the removal of strontium and the total rare earths from large volumes of low-level waste. Cesium can be removed from these wastes by the addition of clay in the process.

The full potential of this plant in controlling the discharges to White Oak Creek is dependent upon a reliable automatic monitoring and diversion system. The times of increased levels of contamination in the process waste stream are difficult to predict, and, unless all higher levels are selectively diverted for treatment, excessive activities may bypass the plant. During the first five months of operation the automatic control valve was not in service, and the plant was operated only five days a week. Sixty-six per cent of the waste was bypassed to the Creek on a predetermined time schedule. Analyses of samples during this preliminary period indicate that the reductions of radioactivity in discharges to the Creek were not as great as should have resulted if all of the higher levels of waste had been treated before discharge.

The experience in operation to date indicates that this facility will fulfill the primary purpose of increasing the factor of safety in the control of radioactive wastes released to the Clinch River. Designed

especially for the removal of strontium, the plant has shown reductions of 80 to 90% of the strontium in the wastes that have been treated. The other radionuclides that have a significant, though less dominant, influence on the MPC values for water can be reduced 50 to 60%, or more. Finally, and of even greater importance, the treatment plant provides a system for detention and treatment of accidental or emergency releases which might exceed the capacity of the river.

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Representative HOLIFIELD. Our next witnesses are Mr. J. R. Horan of AEC, Idaho, and Dr. R. L. Doan of the Phillips Co., who have waste management operations at Idaho.

I will ask both you gentlemen to come forward if you will. You may lead off, Mr. Horan.

STATEMENT OF JOHN R. HORAN,[2] DIRECTOR, HEALTH AND SAFETY DIVISION, IDAHO OPERATIONS OFFICE, U.S. ATOMIC ENERGY COMMISSION

Mr. HORAN. Mr. Chairman, members of the committee, first I would like to express my appreciation for the opportunity to present this information. I should like to confine my testimony primarily to the methods and procedures for control and disposal of radioactive waste material at the National Reactor Testing Station.

The National Reactor Testing Station was established in May 1949 as a place where U.S. Atomic Energy Commission could build, test, and operate various types of nuclear reactors, allied facilities and equipment, with maximum impunity for all concerned. The station is administered by Idaho Operations Office at Idaho Falls, Idaho. Six major operating contractors are engaged in activities on the 572,000-acre reservation, as part of the Commission's reactor development program. These contractors are Phillips Petroleum Co., Westinghouse Electric Corp., General Electric Co., Argonne National Laboratory, Atomics International, and Combustion Engineering, Inc. The completed program function to date embraces 24 installations, including 17 operable reactors and a chemical processing plant; while 9 additional installations, including 5 reactors, are under construction.

Overall responsibility for the health and safety of the people, property and livestock in the general area of the NRTS is vested in the Manager of Idaho Operations Office. The Director, Health and Safety Division, is delegated the general responsibility to insure that the NRTS operations are conducted in such a manner as to prevent undue risk to personnel and property on or off the station, and to assure compliance with Government codes and regulations pertaining to waste disposal in accordance with established permissible levels of radiation exposure.

Senator HICKENLOOPER. I wonder what statement Mr. Horan is reading from. I cannot find any of these words in the one I have here.

Representative HOLIFIELD. Did you submit to us copies of your statement, sir?

Mr. HORAN. Yes, Mr. Chairman.

Representative HOLIFIELD. Is this the large one?

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[2] Born: April 3, 1923, Chicago, Ill. Education: B.S. (honors), Loyola University, Chicago, Ill., 1950; graduate study in physics, Vanderbilt University, Nashville, Tenn., U.S. Atomic Energy Commission, Radiological Physics Fellowship, 1950–51; research in nuclear emulsion techniques, Medical Division, Oak Ridge Institute of Nuclear Studies, Oak Ridge, Tenn., 1951–52. Work experience: Senior health physicist, American Cyanamid Co., Chemical Processing Plant (ICPP), National Reactor Testing Station, Idaho, 1952–53; radiation engineer, test engineer, supervisor, industrial hygiene, Westinghouse Corp., Naval Reactor Facility, NRTS, Idaho, 1953–57; director, Health and Safety Division, Idaho Operations Office, U.S. Atomic Energy Commission, Idaho Falls, 1957 to present.

Mr. HORAN. No; the large one is the testimony for the record. Then there is a 6-page copy of the oral testimony.

Representative HOLIFIELD. Apparently we have not received that. At least we don't have it at the table today.

Mr. HORAN. I am sorry for that inconvenience.

Representative HOLIFIELD. It is all right, we are trying to catch up with the general statement.

Mr. HORAN. Mr. Chairman, may I inquire if you have Dr. Doan's written testimony because all three of these were sent at the same time?

Representative HOLIFIELD. We have what I believe is his, but it may not be his summary. It is probably the full statement the same as yours.

Mr. HORAN. I see.

Representative HOLIFIELD. Go ahead.

Mr. HORAN. The operating contractors are responsible for, first, compliance with applicable regulations and waste disposal criteria established by AEC; second, initiating the necessary forms, reports and records required for the NRTS waste disposal program; and third, maintaining a standard operating procedure covering waste disposal. It is the general policy of AEC to dispose of radioactive waste in such a manner that exposure to all personnel is maintained within established permissible limits, and that pollution of the environment does not occur.

The disposal of solid radioactive waste is accomplished by burial or storage only at locations designated by IDO. An 80-acre common burial ground is available near the southwest corner of the NRTS; for low level routine waste, truck shipments to the burial ground are accomplished by a written waste disposal request and authorization. By this means the contractor provides IDO with a description of the waste, the container, and prescribed precautions for handling. For nonroutine disposal of high level material, and this would be material with a direct radiation field at the surface in excess of 5 roentgens per hour, the request and authorization are arranged in advance of the actual shipment so as to minimize personnel exposure.

The basic guide for the disposal of liquid waste is the following: Liquid radioactive wastes discharged to the ground are maintained at levels such that the concentration in water at the nearest point of use down gradient will not exceed one-tenth of the maximum permissible concentration. In the case of a mixture of two or more radioactive isotopes which affect the same part of the body, the maximum permissible concentration of the two or more is the weighted sum of each of the individual isotopes. Thus the exposure potential of a mixture is no greater than the potential of a single radio isotope. In the case of a mixture of isotopes which could affect different parts of the body, the maximum permissible concentration to be used is the one which will give the highest dose to that particular part of the body if ingested in the proportion found in the water under consideration. Solutions which are within the prescribed limits may be discharged to the ground water table through wells, pits, or ponds. Adsorption, dilution, and decay factors determined by IDO may be used in establishing allowable concentrations at points of discharge in order to comply with our basic guide. Solutions which cannot meet the es-

tablished criteria require permanent storage, temporary retention, or further processing.

The decay factor is calculated from considerations of the linear velocity of the subsurface water as well as radioactive decay. The rate of flow in the water table has been estimated by the U.S. Geological Survey from pumping experiments with various wells on the reactor testing station. The average water velocity is about 35 feet per day. Since this cannot be a precise determination, and recognizing that there are variations from location to location, a safety factor of 10 has been incorporated in our calculations. Therefore, we have assumed a linear velocity of 350 feet per day.

The basic guide is implemented by the issuance of standards to NRTS contractors for the release of wastes at the sources and these are presently in effect. They are the following.

First, monthly averages are to be maintained at three times the maximum permissible concentration for water, and these of course are the values listed in the National Bureau of Standards Handbook No. 52.

Second, maximum levels of release for any period are to be maintained at 10 times the maximum permissible concentration in water.

Third, at any time that effluent released at concentrations greater than three times the maximum permissible concentration for water, immediate notification of the director of the IDO Health and Safety Division is required.

Only one relaxation of the above standard has been permitted. On April 16, 1958, the chemical processing plant was permitted to begin discharging at concentrations of the order of 100 times the maximum permissible concentration in water for 1 specific isotope, namely, iodine 131. This release was to be averaged over a 24-hour period. The cost of concentrating this type of waste for storage had become excessive, more than $100,000 per year. Following this revision, additional surveillance of monitoring wells at a distances of 700 to 900 feet downstream revealed no change in normal background levels. We are convinced that this economy has been achieved without increased risk to NRTS personnel or to the general public.

Airborne radioactive waste discharged to the atmosphere are controlled to insure that overexposure of personnel does not occur, either at the NRTS or in its environs, as a result of direct radiation exposure, inhalation, or ingestion of radioactive materials emanating from the operations at NRTS. Specific regulations are intentionally general and flexible in order that maximum operating time may be afforded to the various plants.

Whenever there is a planned or actual release of any radioactive material which may result in an exposure of 25 millirem to the critical organs of personnel beyond the perimeter of a project area, notification of the Health and Safety Division is mandatory. All planned releases are conducted under IDO surveillance and monitoring, and are subject to meteorological control based on data and recommendations of the U.S. Weather Bureau.

The release of radioactive effluent to the atmosphere is permitted whenever favorable diffusion conditions exist. Should the exposure resulting from operations emitting airborne waste reach the annual

maximum permissible value of five-tenths of a rem per year to the body, three rem to the thyroid or skin, or 1.5 rem for other organs, no further radioactive effluent from the NRTS will be permitted over the affected area for the remainder of the period concerned. If the operation of any particular facility has contributed an equitable share of the dose permitted to any particular area or sector, IDO may restrict future operations by that facility. In addition, the Director of the Health and Safety Division is authorized to order the immediate shutdown of any operation which is considered hazardous to the health and safety of people, property, livestock, et cetera, on the NRTS and its environs.

Any facility releasing radioactive waste to the environment is required to submit a monthly waste report to IDO. This report provides the official record of the type and quantity of waste disposed of and is used for the purpose of research, control, and audit of the disposal procedures.

Our Health and Safety Division coordinates cooperative research programs with other Federal agencies who are recognized authorities in various phases of waste disposal. These agencies are the U.S. Weather Bureau, the U.S. Public Health Service, and the U.S. Geological Survey. Each group is performing basic and applied research in such fields as micrometerology and atmospheric diffusion, hydrology and ion-exchange as well as effluent monitoring techniques, so that the NRTS can pursue a realistic course in radioactive waste disposal which is necessarily balanced between two extremes, hazard to the general public on the one hand and prohibitive operational expenses on the other.

In conclusion, we are confident that we have developed a realistic program of waste management. NRTS waste disposal operations have not resulted in any significant exposure to personnel or property on or off the National Reactor Testing Station.

Representative HOLIFIELD. Thank you, sir.

May I ask you, in relation to some of these charts that you have presented, on page 8 I see the amount of solid radioactive waste disposed of at the National Reactor Testing Station has gone up from 70 curies in 1952 to 15,000 curies in 1957 and 10,000 curies in 1958. That gives you an idea of the rate this problem is advancing, does it not?

Mr. HORAN. That is right. Of course, these figures reflect the growth of activity at the National Reactor Testing Station.

Representative HOLIFIELD. On page 9, referring to the capital costs, am I right in assuming that the total capital costs for the disposal of liquids, gases, and solids is $24,938,700 to date, or is that annual?

Mr. HORAN. That is to date. The annual cost is approximately $1½ million.

Representative HOLIFIELD. Now, that does not include the Material Testing Reactor and the ETR and ICPP operated by the Phillips people? That involves another capital cost of about $9 million, is that right?

Mr. HORAN. No, Mr. Chairman. Page 9 is a summary of the entire cost at the National Reactor Testing Station by all contractors. Then the following pages we have the breakdown by the various facilities.

Chairman DURHAM. Are you receiving any material from any other operations except what is going on at Idaho?

Mr. HORAN. Yes, sir; we do. We receive a large volume of waste from the Rocky Flats plant near Denver, Colo., but this is the only other contributor of waste at our location.

Representative HOLIFIELD. Is that a processing plant?

Mr. HORAN. It is a weapons fabrication facility.

Representative HOLIFIELD. Are there any further questions of Mr. Horan?

Chairman DURHAM. You say in your statement you are going to build nine more reactors there. Five are under contract, I believe, at the present time; is that correct?

Mr. HORAN. Five under construction; yes, sir.

Chairman DURHAM. That problem is going to increase instead of decrease?

Mr. HORAN. That is correct. We do anticipate an increase in the quantity of waste to be disposed for each facility that is completed.

Chairman DURHAM. As I understood your statement, your disposal of radiation material is very similar to that used at Oak Ridge; you have deep wells.

Mr. HORAN. We do use injection wells.

Chairman DURHAM. How long have you been operating with the deep-well system?

Mr. HORAN. Since 1953.

Chairman DURHAM. Do you mind giving us the depth of those wells?

Mr. HORAN. The well is approximately 600 feet deep.

Chairman DURHAM. You have no seepage problem at all into the water flow?

Mr. HORAN. We actually have a total of 15 monitoring wells extending for distances of 700 feet down the water gradient. The furthest one that we have is approximately a mile from the disposal well. We have detected no increase above background on any of these wells to date.

Chairman DURHAM. How are you disposing of your solid wastes?

Mr. HORAN. This is by burial in the southwest corner of the National Reactor Testing Station, and of course the burial is below the surface of the soil. The water table at this particular location is at a depth of approximately 750 feet.

Representative HOLIFIELD. Therefore, there is little chance for any seepage from your burial ground to the water level?

Mr. HORAN. That is correct. We are also located in a desert area with minimum rainfall, and of course the majority of the rainfall does disappear by evaporation rather than by seepage.

Chairman DURHAM. One great advantage out there is that you have such a large acreage, 500,000 acres or more.

Mr. HORAN. Yes, sir; that is correct.

Representative HOLIFIELD. Senator Hickenlooper?

Senator HICKENLOOPER. I would like to ask Mr. Horan about this cost summary. That is your cost at this one station, is it not?

Mr. HORAN. That is correct, sir.

Senator HICKENLOOPER. This is not the cost of the entire waste disposal program in the United States?

Mr. HORAN. No, sir. This is restricted to the National Reactor Testing Station.

Senator HICKENLOOPER. In Idaho?

Mr. HORAN. That is right.

Senator HICKENLOOPER. Now, do you get waste materials from other places to dispose of there?

Mr. HORAN. Only the Rocky Flats plant.

Senator HICKENLOOPER. Just the Rocky Flats plant?

Mr. HORAN. Yes.

Senator HICKENLOOPER. Do these costs include the cost of transportation and handling of that waste before it gets to you or is it just f.o.b. Idaho?

Mr. HORAN. Rocky Flats does pay for the costs that are involved in the disposal at our burial ground, so the charges are reflected to them.

Senator HICKENLOOPER. The charges are reflected to them, they pay the transportation?

Mr. HORAN. Yes, sir.

Senator HICKENLOOPER. The cost of transportation of these wastes is a very substantial amount, with the care that has to be used?

Mr. HORAN. Actually, the material that we receive from Rocky Flats is low activity, so that basically it can be put aboard a conventional truck without shielding. So it is no more expensive than a shipment of the same quantity of other material from the same plant.

Senator HICKENLOOPER. That is all.

Representative HOLIFIELD. Senator Pastore.

Senator PASTORE. For how long will that burial ground be considered quarantined?

Mr. HORAN. Indefinitely.

Representative HOLIFIELD. Are there further questions? If not, thank you very much, Mr. Horan.

(The formal statement of Mr. Horan follows:)

RADIOACTIVE WASTE DISPOSAL MANAGEMENT AT THE NATIONAL REACTOR TESTING STATION

Statement submitted for the record by Mr. John R. Horan, Director, Health and Safety Division, Idaho Operations Office, U.S. Atomic Energy Commission for presentation on January 29, 1959, at the public hearings on industrial radioactive waste disposal, session III–c of the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy

GENERAL

The National Reactor Testing Station (NRTS) was established in 1949 as an area where the U.S. Atomic Energy Commission could build, test and operate various types of nuclear reactors and allied plants and equipment with maximum impunity for all concerned. The station is administered by the Idaho Operations Office of the U.S. Atomic Energy Commission, which has offices in Idaho Falls, Idaho.

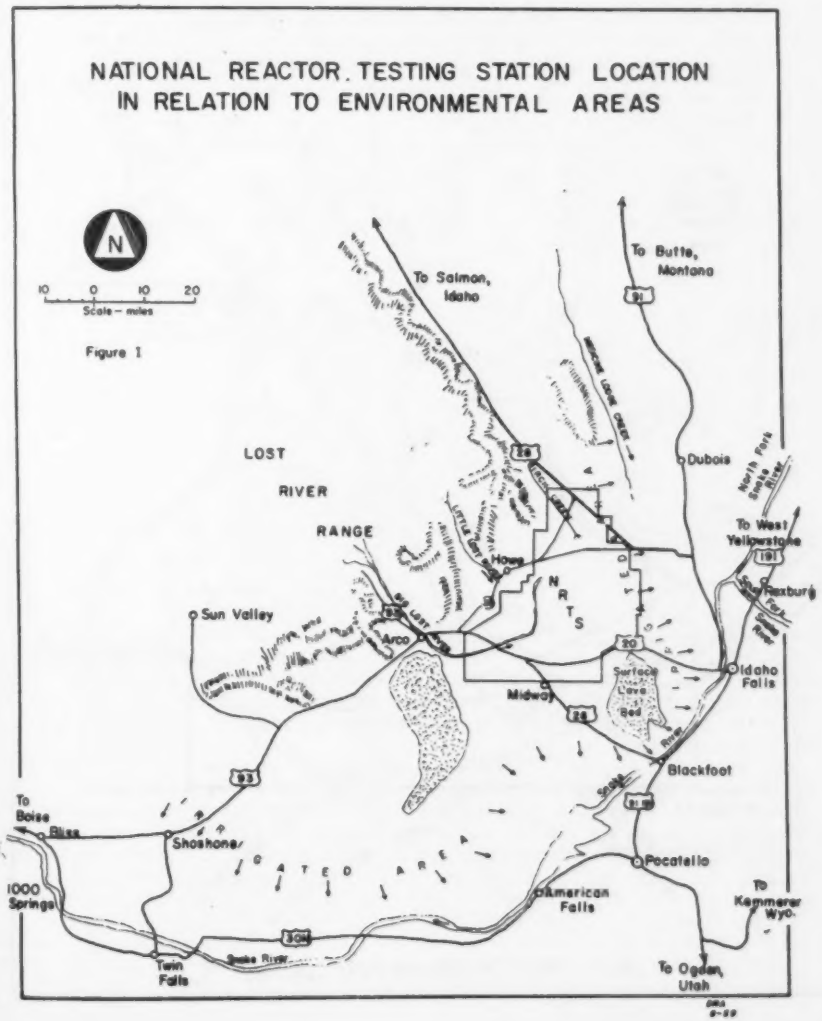
Centered on a former Naval Ordnance Plant Proving Ground, the NRTS covers some 894 square miles of desert land on the Snake River Plain in southeastern Idaho. Its size, enables scientists and engineers engaged in nuclear research to carry on reactor experiments vital to the atomic energy reactor development program without undue hazard to the public. Figure I illustrates the location of the NRTS in relation to the surrounding environment.

At one time in the geologic past the area, which the plain now occupies, was a large depression. This depression was progressively filled by flows of lava which forced the river south to its present location. Voids or openings occurred between each flow or lava. Other openings were caused by gas, shrinkage, differential movement, interbeds of cinders, scoria, and broken or fractured lava

formations. Beds of soil material also were enclosed, and these openings along with interbedded zones became filled with water, forming a water table. At the present time the depth to the water table ranges from 200 feet below the land surface in the northern section of the NRTS to 800 feet below the land surface in the southern section. This unconfined zone of saturation is often compared to an underground lake. It is unique in that it is the largest such occurrence of this nature in the Nation.

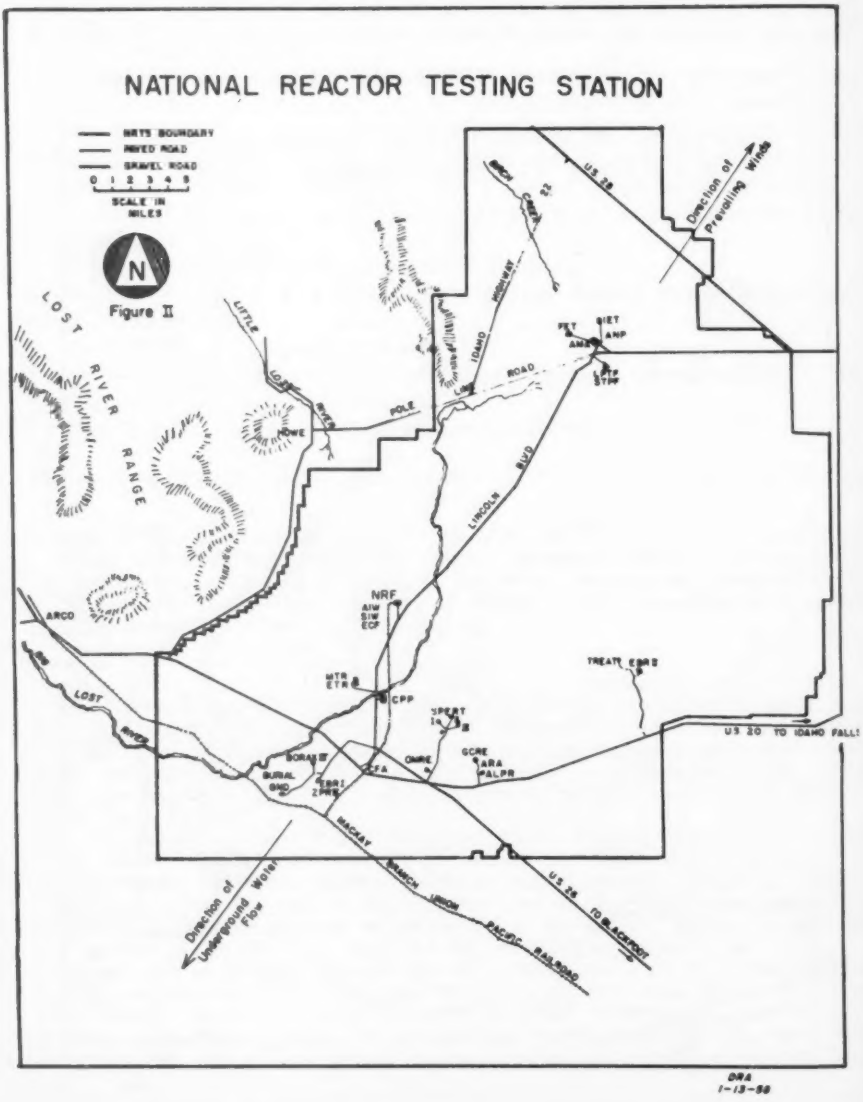
This water table is supplied by underflow from the adjacent mountainous area to the north, northwest, and northeast, and from infiltration along the streams which arise in these areas, including the Snake River. Irrigation water which is not required by plants, retained by the soil, nor lost be evaporation also enters this underground system.

The underground water has a southerly or southwesterly gradient and consequently flows in that direction. Springs which are supplied from this system emerge about 100 miles to the southwest in the Snake River Canyon between Twin Falls and Bliss along a front of about 40 miles. These springs contribute about 8,000 cubic feet per second to the flow of the Snake River. This water table also serves as the source of supply for all consumptive use of water at



the NRTS and area to the southwest, and as a disposal and diffusion place for low level radioactive liquid waste.

Figure II shows the various installations and facilities at the NRTS in relation to the direction of movement of the wind and the underground water. The prevailing winds are from the southwest. These winds are most frequent in the daytime and sometimes persist for periods of 24 hours or longer. At night, due to the intense radiational cooling of the desert floor, the winds usually flow from the northeast. About 75 percent of the winds blow from the southwest or northeast quadrant. Figure III shows the percentage of the time that the wind blows from the various compass directions at different NRTS locations together with velocity. Figure IV shows the relationship of population density to wind direction. The relative consistency of these wind directions allows location of reactors and stacks in order to minimize the probability that winds will carry undiluted effluent over nearby inhabited installations and also allows planning of controlled releases of radioactivity to the atmosphere.



The climate of the area is semiarid with a low percent of cloudiness. The mean annual rainfall is 7.9 inches. Little frontal type weather occurs between May and October creating a great similarity in weather during this period. Severe weather such as tornados, blizzards, duststorms, etc. are not a problem.

The station is under the general supervision of the Commission's Idaho Operations Office. Testing and development of reactors is conducted by operating contractors, of which there are six at the present time. The Idaho Operations Office establishes the policies and procedures for disposal of radioactive materials. It is the policy of the AEC to dispose of radioactive materials in such a manner that exposure to operating personnel and the public is maintained within internationally established limits.

The general responsibility for waste disposal operations is delegated by the IDO Manager, to the Director, Health and Safety Division. This Division conducts research, develops detailed plans and procedures, inspects, supervises, monitors, and determines that the necessary control measures are applied by the contractors. The burial ground for disposal of solid waste is also managed by this Division.

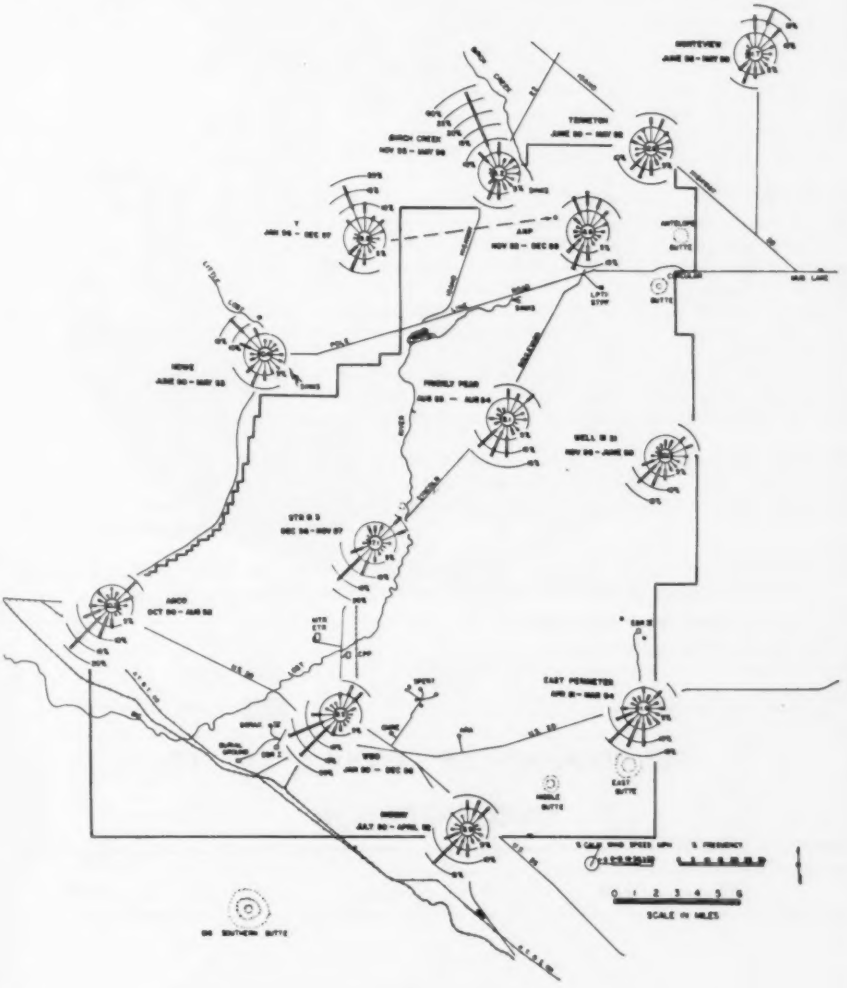
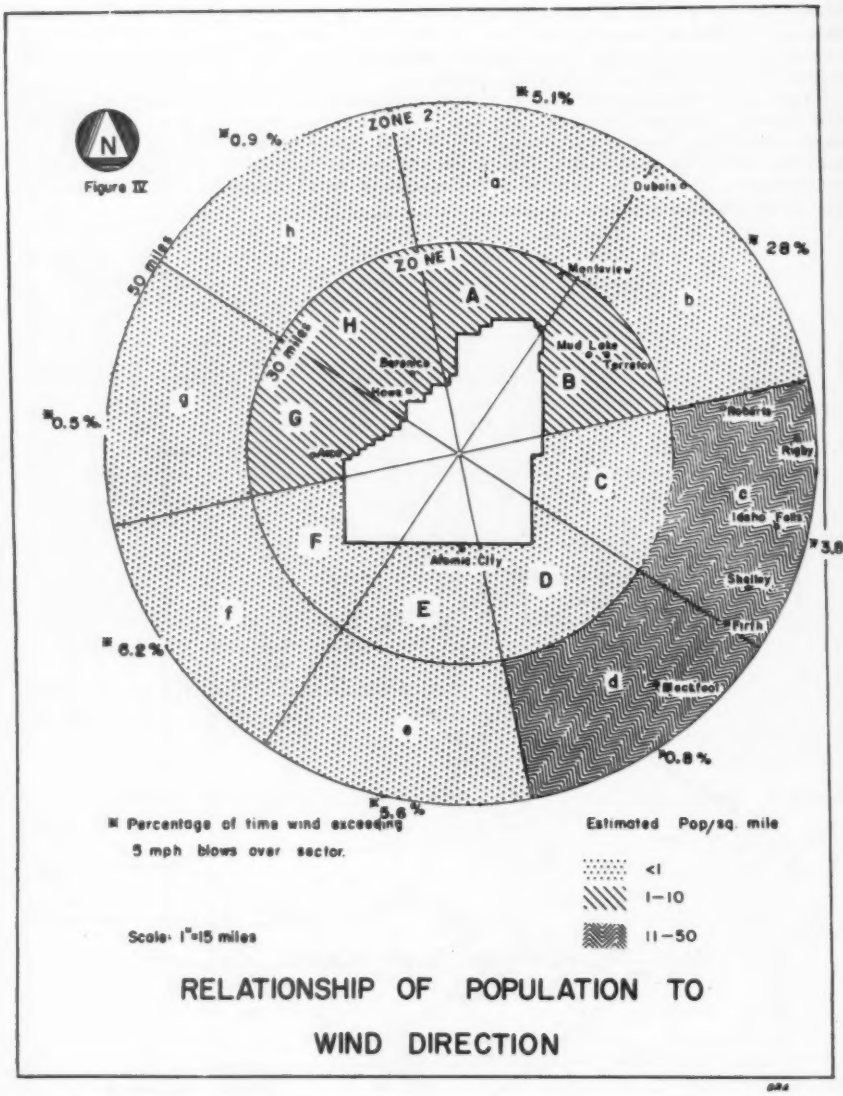


FIG. III ANNUAL WIND ROSES 20 FOOT LEVEL



RELATIONSHIP OF POPULATION TO WIND DIRECTION

DISPOSAL OF LOW LEVEL LIQUID WASTE

Liquid radioactive wastes which are produced as a result of operations in the various plants at the NRTS are monitored. High level material is retained in permanent storage vessels, while those which can be safely discharged to the environment are disposed of in the underground water. This disposal presents no hazard, and results in enormous economic savings by eliminating the need for processing and storage. Disposal is accomplished by discharging directly through disposal wells which penetrate the water table, or indirectly through seepage pits or ponds.

A number of uncertainties exist in regard to the hydrology of this area, such as the exact route and rate of movement of the underground water. Limitations on the concentration of radioactive waste which can be released to the water table are stringent, and are based on cautious assumptions in order to insure that safe concentrations are not exceeded. For example, it is estimated that the underground water moves at a rate of 35 feet per day. In calculating the amount of activity which will remain at some downstream location after radioactive decay, the rate of movement is assumed to be 350 feet per day thus allowing a safety factor of 10. One-tenth of the maximum permissible level for the continuous consumption of water is considered the maximum allowable level for any radioactive isotope in water which is to be used for consumption. In order to assure that the levels do not exceed this amount a monitoring program is conducted, which involves sampling of all production wells on the NRTS as well as approximately 25 wells scattered across the plain south of the boundary. Radioactivity levels above normal background fluctuations have not been detected at any distance downstream from operating areas as a result of disposal operations.

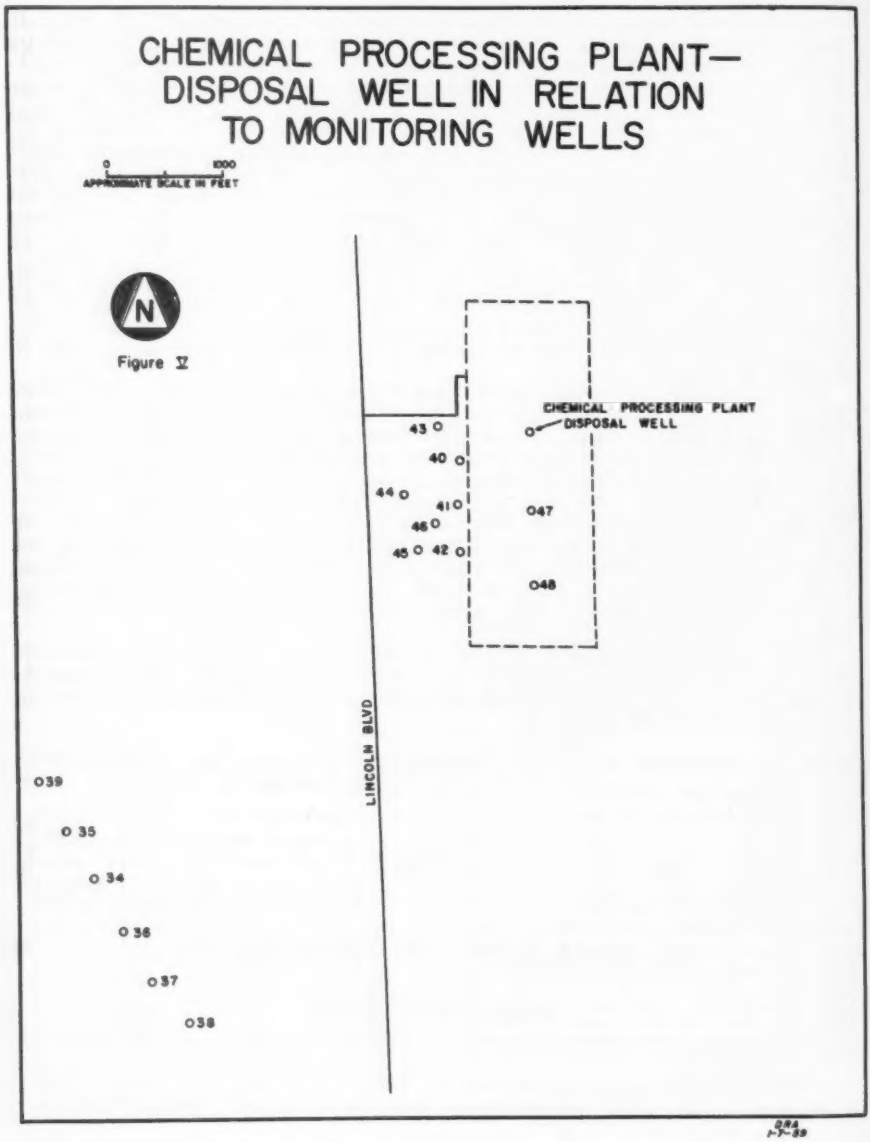
A joint research program with Ground Water Division of the U.S. Geological Survey is being conducted in the vicinity of the chemical processing plant which discharges a large amount of low level radioactive waste through a disposal well. This program is for the purpose of establishing flow patterns and rates of movement and involves 15 wells, as shown by figure V, drilled at distances varying from 700 to 6,000 feet downstream from the disposal well. Samples are periodically collected from these wells and analyzed for radioactivity. As a means of determining the rate of flow and delineating the flow pattern, the samples are also analyzed for the content of common salt which is discharged at the rate of approximately 2 tons per day from water softening processes. In addition fluorescein dye and special radioactive tracers are used in this program.

Table I gives the estimated amount of radioactivity which has been released to the ground water at the NRTS from 1952 through 1958. The percentage composition has been broken down according to the radioactive half lives of the isotopes in the effluent.

TABLE I.—*Estimated amount of radioactivity released to the underground at National Reactor Testing Station*

Year	Amount in curies	Percentage composition by half life		
		Less than 10 days	10 to 100 days	More than 100 days
1952	75	100	0	0
1953	240	92	7	1
1954	1,150	99	0	1
1955	1,265	98	1	1
1956	870	75	20	5
1957	1,700	80	10	10
1958	3,600	75	20	5
Total	8,900	85	10	5

This activity has been diluted with plant process water before disposal so that the concentration has at all times been maintained at safe levels. During 1958 approximately 650 million gallons of water were used in making this dilution. Approximately 85 percent of the material discharged had a half life of less than 10 days. Assuming that this material had a specific half life of 10 days there would only be one-tenth of 1 percent of the activity remaining after a period of 3 months. This considers the effect of only one factor, physical decay, and disregards other modifying factors such as dilution and adsorption.



DISPOSAL OF RADIOACTIVE WASTE TO THE ATMOSPHERE

The evaluation and control of the hazards presented by the discharge of airborne radioactive waste is initiated during the planning and design stages of a new reactor or associated installation. Design criteria, operational procedures, and the proposed locations are studies to make certain that the released airborne radioactive waste will be maintained below maximum permissible concentrations.

Radioactive gases are filtered to remove particulate material prior to release through stacks equipped with monitoring instruments to determine the rate of discharge and the characteristics of the material. The predominant material released is in the form of an inert gas, specifically Argon 41, which will not combine with body chemicals and produces only direct exposure and consequently does not contribute significantly to the waste disposal problem.

The recovery of radioactive barium from fresh fuel elements at the chemical processing plant discharges iodine 131. This isotope is of potential concern to the station and its environment because of the affinity of human and animal thyroids. Releases of this material are made only after a thorough review of the proposed operation and are made to coincide with favorable diffusion conditions which include wind speeds, directions and temperature variations with altitude. The prime consideration is always minimization of exposure to on-site and off-site personnel and animals.

Criteria are established by the IDO Division of Health and Safety for planned releases which include the recommendations of the U.S. Weather Bureau. During a release the atmosphere is monitored by means of mobile and stationary radiation detection and sampling equipment, in addition to the continuously operating NRTS radiation surveillance network. By these methods deposition and concentrations of released radioactive materials can be defined. The mobile equipment is mounted in trucks or trailers which are positioned in accordance with wind direction. The radiation detection network consists of radiation detection stations located at the NRTS and its surrounding area as shown by figure VI. The monitoring stations are unique in that they provide instantaneous information to a central station by means of a telemetering system. Fallout trays and monitoring film badges are also exposed at these locations. Recently a very satisfactory method of trapping iodine 131 by means of charcoal filters had been developed by the IDO Health and Safety Division and is used where this isotope may be a problem.

Samples of vegetation and small animals are also collected and analyzed in order to determine what effect, if any, the fallout contamination might have on the environment. Wild rabbits furnish a valuable indicator in studying the presence of radioactive iodine on vegetation, since they consume relatively large quantities and are therefore very sensitive monitors. Any iodine which might be on the vegetation accumulates in the thyroid glands of these animals, and measurable amounts have been detected in this manner. Samples of milk from off-site dairy farms also are collected and analyzed as a means of monitoring any radioactive fallout that may be carried beyond the boundary of the station and consumed by grazing animals.

As has been noted, meteorological conditions affecting air movement or conditions of diffusion are of the utmost concern in the control of the discharge of radioactive wastes to the atmosphere. For this reason an operational and research program has been established with the US. Weather Bureau to study, interpret and predict conditions related to the discharge of wastes to the atmosphere. This Bureau maintains a station at the NRTS with a staff of research meteorologists and forecasters. Considerable experience has been gained since the station was established in forecasting weather parameters.

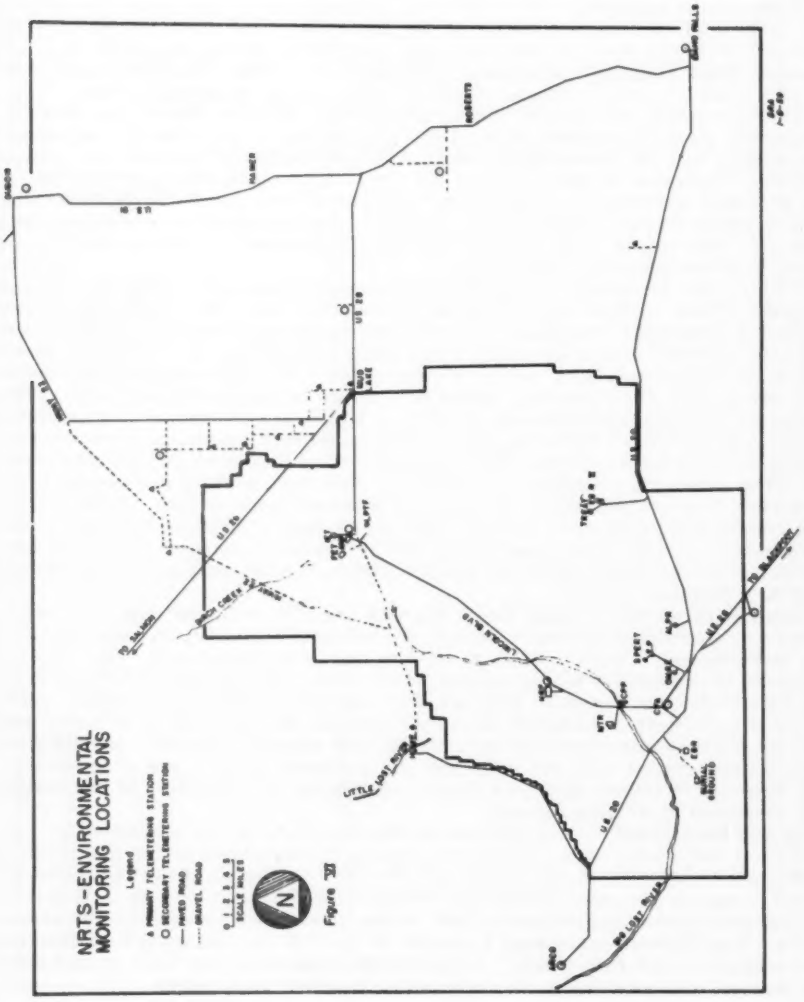

NRTS-ENVIRONMENTAL
MONITORING LOCATIONS
Legend
A PRIMARY TELEMETERING STATION
O SECONDARY TELEMETERING STATION
— PAVED ROAD
---- GRAVEL ROAD
SCALE MILES
N
ARCO
IDAHO FALLS
US 20
US 26
US 93
ATR
CFA
MTR
TRA
ICPP
EBR-II
BORAX
PBF
SPERT
NRF
RWMC
LOST RIVER
BIG LOST RIVER
BIRCH CREEK
MUD LAKE
TO BLACKFOOT
TO ROBERTS

Figure 23

TABLE II.—*Amount of gaseous and particulate waste released to the atmosphere at the National Reactor Testing Station. Over 75 percent of the isotopes have a halflife of less than 2 hours*

Year:	*Amount in curies*
1952	195,000
1953	370,000
1954	271,000
1955	404,000
1956	213,000
1957	77,000
1958	145,000
Total	1,675,000

DISPOSAL OF SOLID WASTE

All solid radioactive waste generated at the National Reactor Testing Station is disposed of in a common burial ground either by dumping into trenches or stacking in large pits. The burial ground consists of an area of approximately 80 acres enclosed by a barbed wire fence. It is located in the southwestern portion of the station and is not directly up the ground water stream from existing or potential plant sites. It is at least 60 miles from the nearest downstream populated area where water may be used. The soil depth varies from 2 to 20 feet. Trenches average 900 feet in length, 10 feet in depth and from 5 to 10 feet in width. Large open pits 6 to 10 feet in depth, 40 to 50 feet in width and 900 feet in length are also used for disposal of low level radioactive materials.

The bulk of the solid waste is low level material consisting mainly of contaminated items such as rags, paper, sample bottles, lumber and metal scraps. These wastes are transported to the burial site in cardboard boxes by means of closed dump trucks where they are placed in pits or trenches.

High level wastes (greater than 5 roentgens/hour at contact) require special handling. Remote handling equipment, shielded containers and remote truck dumping controls are utilized for the disposal of these wastes. This type of material involves metal pieces which have been exposed to high levels of neutron radiation and constitutes a personnel exposure hazard. Such waste is deposited in the lower levels of the trenches and is covered with earth to achieve shielding.

Low level waste from other AEC installations which do not have suitable disposal facilities are shipped to the NRTS for disposal. This waste contributes largely to the total volume of solid waste which has been deposited in the NRTS burial ground.

For inventory purposes, the location of all buried material is recorded in a log. All wastes deposited in the burial ground must be in solid or insoluble form. Corrosion, transfer, and exchange with the soil may occur but due to the arid climate, little if any, water moves through the soil profile. Thus the leaching of this material into the ground water is minimized.

The exact amount of activity and the volume of the material involved is difficult to determine precisely. Table III shows the estimated amount of material which has been disposed of to date.

TABLE III.—*Amount of solid radioactive waste disposed of at the National Reactor Testing Station*

Year	Curies	Volume, cubic yards
1952	70	2,000
1953	800	
1954	1,500	
1955	1,500	2,500
1956	10,000	5,000
1957	15,000	6,500
1958	10,000	9,000
Total	[1] 40,000	25,000

[1] Approximate.

COSTS

The capital cost of the facilities, systems and equipment used in connection with the disposal of radioactive waste, together with the estimated present annual operating and maintenance costs, are shown on the following tables.

The estimated current manpower including administration, technical and operating personnel involved in waste handling operations is also shown.

COST SUMMARY

TABLE IV.—*Estimated cost of disposal of radioactive waste, National Reactor Testing Station*

Item	Type or form of waste			Total
	Liquid	Gaseous	Solid	
Capital costs:				
AEC, IDO Health and Safety	$78,000	$255,000	$30,500	$363,500
MTR, ETR, CPP	19,419,000	2,286,000	134,900	21,839,900
NRF	1,385,000	305,000	85,000	1,775,000
ANP	442,000	467,000	1,300	910,300
Other minor facilities	25,000	25,000		50,000
Total	21,349,000	3,338,000	251,700	24,938,700
Annual operating and maintenance costs:				
AEC, IDO Health and Safety	31,500	178,000	22,000	231,500
CPP	913,615	65,205	14,054	992,874
MTR, ETR (undifferentiated)				116,251
NFR	54,000	11,000	14,000	79,000
ANP	13,400	22,100	2,300	37,800
Other minor facilities (undifferentiated)				25,000
Total				1,482,425
Manpower (man-years):				
AEC			20	
MTR, ETR, CPP			50	
NRF			11	
ANP			5	
Others			3	
Total			89	

TABLE IVa.—*Costs of waste disposal operations at NRTS, IDO, Health and Safety Division and cooperating agencies*

	Type of waste			
	Liquid	Gaseous	Solid	Total
Capital costs	[1] $78,000	[2] $255,000	[3] $30,500	$363,500
Annual operating and maintenance costs	[4] $31,500	[5] $178,000	[6] $22,000	$231,500
Manpower (man years)	2	16	2	20

[1] Monitoring wells, sampling equipment.
[2] Meteorological equipment, telemetering system, mobile monitoring units, communication equipment, sampling equipment, etc.
[3] Fencing and road to burial ground.
[4] AEC: Labor (planning management), sample analysis, indirect costs; USGS: Labor, equipment, materials, supplies, indirect costs.
[5] AEC: Labor, equipment, materials, indirect costs; U.S. Weather Bureau: Labor, equipment, materials, supplies, indirect costs.
[6] Contractor services (Phillips Petroleum Co.), supervision, materials, supplies and indirect costs.

TABLE IVb.—*1. Capital costs of waste disposal systems at MTR, ETR, and ICPP, operated by Phillips Petroleum Co. (P.P. Co.)*

	Type of waste		
	Liquid	Gaseous	Solid
A. Materials testing reactor: Type of service:			
Collection	$555,000	$332,000	
Processing	45,000		
Monitoring	10,000	10,000	
Disposal to permanent storage			$3,700
Total	610,000	342,000	3,700
B. Engineering test reactor: Type of service:			
Collection	474,000	214,000	
Processing	110,000		
Monitoring		50,000	
Disposal to permanent storage			1,200
Total	584,000	264,000	1,200
C. Idaho chemical processing plant: Type of service:			
Collection	3,830,000	1,485,000	80,000
Processing	4,955,000	125,000	50,000
Monitoring	155,000	70,000	
Disposal to permanent storage	9,553,000	250,000	[1]
Total	18,493,000	1,930,000	130,000
D. Total capital cost for disposal systems: Plant:			
Materials testing reactor	610,000	342,000	3,700
Engineering test reactor	584,000	264,000	1,200
Idaho chemical processing plant	18,225,000	1,680,000	130,000
Total	19,419,000	2,286,000	134,900

[1] AEC furnishes.

2. Annual operating and maintenance costs for all process waste disposal at MTR, ETR.

Direct operating costs	$84,240
Indirect costs	32,011
Total	116,251

ANNUAL OPERATING AND MAINTENANCE COSTS FOR COLLECTION, PROCESSING, MONITORING, AND DISPOSAL OF ICPP WASTES

	Direct costs	Indirect costs	Total
A. Collection systems:			
Process waste	$127,440	$48,427	$175,867
Dilute waste	22,788	8,659	31,447
Gaseous waste	15,876	6,033	21,909
Solid waste	10,184	3,870	14,054
Total	176,288	66,989	243,277
B. Processing systems:			
Process waste	190,404	72,354	262,758
Dilute waste	130,680	49,658	180,338
Gaseous waste	5,508	2,093	7,601
Solid waste			
Total	326,592	124,105	450,697
C. Monitoring systems:			
Process waste	87,048	33,078	120,126
Dilute waste	37,260	14,159	51,419
Gaseous waste	20,790	7,900	28,690
Solid waste			
Total	145,098	55,137	200,235
D. Disposal systems:			
Process waste	54,432	20,684	75,116
Dilute waste	11,988	4,556	16,544
Gaseous waste	5,076	1,929	7,005
Solid waste			
Total	71,496	27,169	98,665

	ICPP	MTR-ETR	Total
E. Total annual operating and maintenance costs of CPP, MTR, and ETR.:			
Total annual direct costs	$719,474	$84,240	$803,714
Total annual indirect costs	273,400	32,011	305,411
Total	992,874	116,251	1,109,125

3. Manpower directly involved in waste handling operations (man-years)

	ICPP	MTR-ETR
Direct operating labor	47.0	1.5
Technical assistance	1.4	.5

TABLE IVc.—*Costs of waste disposal at the Naval Reactor Facility—NRTS—operated by Westinghouse Electric Corp.*

	Type of waste			
	Liquid	Gaseous	Solid	Total
Capital costs:				
Collection	$1,135,000		$35,000	$1,170,000
Processing	180,000	$140,000	30,000	350,000
Monitoring	20,000	25,000	10,000	55,000
Disposal	50,000	140,000	10,000	200,000
Total	1,385,000	305,000	85,000	1,775,000
Operating and maintenance costs:				
Collection	32,000		8,000	40,000
Processing	12,000	8,000	5,000	25,000
Monitoring	10,000	3,000	1,000	14,000
Total	54,000	11,000	14,000	79,000
Manpower (man-years):				
Administrative	0.5	0.3	0.2	1
Technical	2	1.25	0.25	3.5
Operations	4	1.5	0.50	6
Total				10.5

TABLE IVd.—*Costs of waste disposal at the Aircraft Nuclear Propulsion Area—NRTS—operated by General Electric Co.*

	Type of waste			
	Liquid	Gaseous	Solid	Total
Capital costs	[1] $442,000	[2] $467,000	[3] $1,300	$910,300
Annual operating and maintenance costs	[4] 13,400	[5] 22,100	[6] 2,300	37,800
Manpower (man-years)	(1.2)	(3.3)	(0.5)	(5.0)

[1] Evaporator building—tank farm, systems.
[2] Stacks, filter, monitors, etc.
[3] Transportation containers.
[4] Processing, monitoring (labor).
[5] Instrument maintenance—sampling, counting, monitoring (labor), indirect costs.
[6] Collection and transportation (labor and equipment).

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Representative HOLIFIELD. Now we hear from your associate, Dr. Doan.

STATEMENT OF R. L. DOAN,[3] PHILLIPS PETROLEUM CO., IDAHO FALLS, IDAHO

Mr. DOAN. Mr. Chairman and members of the committee, I am here to present a statement on behalf of Phillips Petroleum Co., one of the AEC contractors at the National Reactor Testing Station in Idaho, relative to waste management operations program on the collection, handling, processing, and disposal of waste materials at the three plants operated by Phillips; namely, the Chemical Processing Plant, the Materials Testing Reactor, and the Engineering Test Reactor. All of these plants generate radioactive liquid, gaseous and solid waste materials which must be stored or disposed of in such a manner as not to constitute a current or future hazard to people working at the National Reactor Testing Station or to people, animals, and property in areas surrounding the site.

To this end, every reasonable precaution is exercised to insure that materials which must be retained are stored safely and that materials which may be released in prescribed quantities to the atmosphere or to the earth are released under carefully monitored and controlled conditions.

In the case of liquids discharged from the plants and returned to the earth via leaching ponds or disposal wells, the average concentration of radioactive material in the liquids is kept within limitations prescribed by the AEC, which are based on information contained in National Bureau of Standards Handbook 52. The same is true of waste gases discharged to the air. Constant monitoring outside the perimeter fences of the plants, conducted by the site survey branch of the Idaho Operations Office Division of Health and Safety, insures that the rates of radioactive gaseous waste discharge, together with prevailing meteorological conditions, do not give rise to bulk air contamination which transcends established limits.

Radioactive solid waste materials are not subject to discharge rate control. These materials are buried in a burial ground operated and controlled by the Health and Safety Division of the Idaho Operations Office. Solid materials which are insoluble in water are buried directly

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[3] Born September 7, 1898, at Lapel, Ind. Educated in Lapel public schools. Graduated from Indiana University with A.B. degree in 1922 and M.A. in 1923, both in physics. Taught in grade schools and high schools of Indiana and Illinois and at Reed College in Portland, Oreg. Received Ph. D. in physics from the University of Chicago in 1926.

Spent 7 years as research and development engineer with Western Electric Co. (1926–33) and 3 years in X-ray and cosmic ray research at the University of Chicago (1933–36) before joining Phillips Petroleum Co. research department in 1936. On civil leave of absence from Phillips during the war period, served as first director (January 1942–August 1943) of the University of Chicago's plutonium-oriented metallurgical laboratory (now Argonne National Laboratory), and from August 1943 to October 1945 as first director of research at Clinton Laboratories in Oak Ridge (now Oak Ridge National Laboratory). Returned to Phillips as associate director of research in October 1945 and became director of research in 1950. From 1951 to present, manager of Phillips Atomic Energy Division, whose activities are devoted exclusively to work at the National Reactor Testing Station in Idaho under contract with the Atomic Energy Commission. Member of Advisory Committee on Reactor Safeguards.

in the ground, while soluble materials are suitably packaged in waterproof containers before burial.

Since the three plants under the jurisdiction of Phillips Petroleum Co. do not have identical problems of waste handling, each unit will be treated separately, starting with the Idaho Chemical Processing plant which generates the largest quantity of radioactive waste materials.

Idaho Chemical Processing Plant: The ICPP is devoted principally to the recovery of enriched uranium from spent reactor fuel elements. The recovery process involves dissolution of the fuel elements in acid followed by extraction of essentially all of the enriched uranium from the resulting solutions. The reprocessing operation generates a liquid waste containing most of the nonvolatile fission products produced during reactor operation and a gaseous waste containing radioactive krypton and iodine.

Liquids: The principal sources of radioactive liquid wastes are the three cycles of liquid—liquid extraction by which the enriched uranium is recovered from the dissolved fuel elements. The aqueous waste material from the first step contains the greatest quantity of fission products while the waste from the third cycle contains the least. Because of the quantities of contained fission products, all of these wastes are stored for an indefinite period in 300,000 gallon underground stainless steel tanks.

Due to the quantities of heat-generating fission products in the first cycle waste, it is necessary to store this material for several years in water-cooled tanks to eliminate boiling and reduce corrosion of tank materials. The second and third cycle wastes are stored together in uncooled tanks. The volume of waste stored varies from 50 to 150 gallons per pound of uranium recovered, depending upon the uranium content of the fuel being processed.

As the waste material flows from the extraction columns, it is temporarily collected in small storage vessels to permit analysis for residual uranium. If the uranium content is satisfactorily low, the waste is fed to evaporators where part of the water is removed to effect volume reduction with resultant economy in storage space. From the evaporators the waste is transferred directly to the appropriate underground storage tanks. The tanks are so installed that a leak can be detected immediately.

In the event of a leak, the contents of the tank would be transferred to an empty tank which is always available to meet such an emergency. There have been no leaks to date.

Secondary sources of radioactive liquid wastes are the process equipment cell floor drains, some laboratory drains, equipment decontamination solutions and the water removed from the primary process wastes during partial evaporation. These wastes are normally low in activity but above levels which can be returned directly to the area water table. Waste materials from these sources are collected in small underground tanks, sampled for uranium content, and then fed to an evaporator where most of the water is taken overhead at an activity level low enough to permit disposal, after reasonable dilution, directly to the area water table by means of a disposal well.

The bulk of the fission products which enter the evaporator with the feed stream remain in the kettle product and are transferred to permanent storage in the large underground tanks.

All liquid discarded in the disposal well is continuously sampled in order to provide a daily record of total radioactivity placed in the water table. In addition, instruments constantly monitor the stream to the well and sound an alarm in the event the activity reaches a preset level. The average level of activity discharged to the water table is held within the limits established by the Idaho Operations Office of the Atomic Energy Commission.

The collection, processing, monitoring and storage of liquid wastes are costly items, both from an investment and operating viewpoint. The collection of these wastes, including the necessary piping, intermediate storage tanks, sampling systems, transfer jets, instrumentation and housing, requires an investment of $3,830,000 at the ICPP. The evaporators, piping, sampling system, instrumentation and housing incidental to processing the liquid wastes for storage represent another $4,955,000 investment. Cost of monitoring equipment is $155,000.

Finally, the total investment in storage tanks is $7,680,000, made up as follows: Nine 300,000-gallon permanent storage tanks for aluminum waste of which six are cooled. Average cost is $789,000, which is equivalent to a storage cost of $2.63 per gallon, including apportioned piping. Four 30-thousand gallon tanks are provided for storage of zirconium waste.

Senator HICKENLOOPER. $2.63 per gallon for a year?

Mr. DOAN. This is the total investment cost in the storage. Then there are four 30,000-gallon tanks provided for storage of zirconium waste at total cost of $580,000, which is equivalent to a per-gallon investment cost of $4.90 per gallon.

During dissolution of the fuel elements, some gaseous radioactive fission products are released inside the dissolver vessels. These gases are removed under vacuum through a dissolver off-gas system which retains some of them, and the residue is filtered to take out radioactive particulate matter, diluted with air and then discharged to the atmosphere through a 250-foot high stack. For normal plant operation, the dilution afforded by plant ventilation air, plus the additional dilution effected by dispersal to the atmosphere from the top of the stack, yields an activity level at the point of exposure which meets established limits.

In a separate process for the recovery of radioactive barium, large quantities of radioactive iodine and xenon are contained in the evolved waste gases. The reason for that is that in the special process instead of allowing the fuel to be cooled for 3 or 4 or more months as it is in the case of normal processing, it is allowed to cool only a few days.

The quantities and release rates of these volatile materials are such that except under very favorable meteorological conditions it is not permissible to release them directly to the stack. In order to stay within prescribed limits the gases containing iodine and xenon are scrubbed with caustic solution to partially absorb iodine, passed through activated charcoal to further reduce the iodine content, and, if required by meteorological conditions, can be partially retained in

a 10,000-cubic-foot gasholder for gradual release to the atmosphere under favorable weather conditions.

During these operations which are conducted on an average of 1 day each month, constant liaison by radio exists between operations personnel responsible for the process and representatives of the U.S. Weather Bureau and the Health and Safety Division of the Idaho Operations Office, to insure that the quantities of iodine and xenon released do not constitute a hazard within or outside the National Reactor Testing Station.

The total investment in off-gas collection equipment at the ICPP is $1,235,000, which includes the vessel off-gas system, the dissolver off-gas system, and incidental instrumentation. There is also a $125,000 filter system, and an investment of $250,000 for the stack, stack heater, and duct work, with an additional $70,000 allocated for improvement in stack gas monitoring.

Minor amounts of radioactive solid waste materials require disposal from the ICPP. These materials which comprise worn process equipment, rags and mops used in decontaminating equipment, and other small items, are appropriately packaged and hauled to a burial ground for permanent disposal. Decontamination and disposal equipment for handling these solid wastes is valued at $30,000.

Approximately 50 employees at the ICPP are engaged full time in handling various phases of the collection, processing, monitoring, and disposal of liquid, gaseous, and solid radioactive wastes. The total annual operating and maintenance costs are $919,580, made up of $666,180 direct and $253,700 indirect expense.

Materials testing reactor: The materials testing reactor is a high flux, light watercooled and moderated reactor designed to accommodate a large variety of experiments ranging from small capsules to large high temperature high pressure equipment rivaling in complexity the reactor itself.

The fuel elements, control rods, and other equipment surrounding the reactor core are cooled by high purity water, as are many experiments in and surrounding the core.

A graphite cube and heavy steel thermal shield which surround the reactor tank are cooled by air which is drawn through the graphite from the reactor building and subsequently discharged to the atmosphere.

This is slightly different from the air-cooled graphite reactor Dr. Morgan talked about awhile ago because we have no fuel within the reactor. In this particular case, it is outside the tank.

Operation of the reactor and the various experiments gives rise to liquid, gaseous, and solid wastes which are radioactive and must, therefore, be disposed of in a manner consistent with established health physics requirements.

The air and water which cool the reactor become radioactive due to neutron bombardment.

Various experiments may from time to time give rise to liquid and gaseous waste which must be disposed of along with the principal waste from the reactor.

Solid wastes are principally sections of experimental equipment and portions of fuel elements and control rods which become radioactive while in the neutron atmosphere of the reactor.

Investment is $610,000 for liquid collection, processing, monitoring, and disposal; $342,000 for gas collection, monitoring, and disposal; and $3,700 for solid disposal.

Annual operating and maintenance costs for MTR and ETR are $78,000 direct costs, and $30,400 indirect.

Liquids: The bulk of radioactive liquid waste generated at the MTR emanates from the primary process water system used to cool the reactor core components. While this water is demineralized to high purity prior to use, it still contains residual traces of minerals, principally sodium 23, which upon passing through the core of the reactor becomes radioactive.

In addition, small quantities of other elements are picked up by the water from the piping, the reactor tank, and core components and experiments installed in and around the core; these elements also become radioactive and add to the total activity in the water system.

Occasionally, an experiment which is being cooled by the process water may rupture or otherwise leak with a resultant release of fission products or activated elements directly to the cooling water.

Thus, we have three sources of activity in the primary cooling system, with activated sodium normally contributing the major portion of activity.

In order to prevent the gradual buildup of trace elements in the primary cooling water, a portion of the stream is constantly bled off and replaced with fresh water. The water removed from the system is fed to one of the two 350,000-gallon concrete retention basins where the average time required to flow from inlet to outlet is equivalent in decay time to 10 half lives of the radioactive sodium.

By the time the water has spent 150 hours in the retention basin, the activity has reduced to the point where the water may be returned to the ground by means of a leaching pond.

On occasions, when the activity in the process water stream has been raised by a leaking experiment to the point where normal disposal cannot be effected, a bypass demineralizer is employed to continuously demineralize a portion of the primary water until the total activity is reduced to a level which can be safely handled by normal disposal procedures.

The demineralizer resin is capable of removing from the water most fission products and source materials which can be added to the water from experiment failures. When the resin reaches its capacity for adsorption of these materials, it is removed in sealed steel tanks and buried with other solid wastes.

A secondary source of radioactive liquid waste is a system which collects warm and hot wastes from laboratory drains, experiment installations and floor drains on and around the reactor. This system collects the waste in small catch tanks where it is held until the major contaminants are determined and a measurement made of the activity level.

If the activity level and contaminant content are such that the waste can be discharged to the water table, it is pumped to the previously mentioned retention basin for disposal along with the purge stream from the reactor.

In the event the activity level of the waste in a catch tank is too high for direct disposal, it is transferred to one of two 10,000-gallon

storage tanks to await treatment for removal of radioactive materials by ion exchange in one of four demineralizers at the Engineering Test Reactor.

From the demineralizers the cleaned waste is returned to ground via the retention basin and leaching pond.

Gases: The cooling air which removes heat from the thermal shield and graphite in the MTR becomes radioactive as it passes through the reactor due to activation of argon which is a minor constituent of air.

This air flows through the reactor at the rate of 24,000 cubic feet per minute and is discharged to the atmosphere through a 250-foot-high stack. The effluent from the stack is constantly monitored to yield information on activity level and total activity discharged.

In addition to the radioactive air from the reactor cooling system, there are contributions to the total made by the reactor cooling water degassing system, certain experiments and laboratory hood exhausts. These streams are combined with the reactor cooling air exhaust and discharged through the stack.

In the event of experiment rupture in the reactor tank, the activity contributed by the cooling water degassing system may exceed that from argon activation in the cooling air.

However, due to the use of a bypass demineralizer on the cooling water stream, this contribution is short lived and to date has not created a disposal problem.

Solids: Radioactive solid materials which require disposal are:

1. Fuel element and control rod end boxes, which are removed prior to reprocessing to save cost; and
2. Parts of experimental equipment, which has served its purpose but has become radioactive due to exposure in the reactor. These materials normally would be looked upon as scrap in a nonatomic energy industry, but because of the induced activity cannot be handled as scrap through industrial channels.

It is normal practice to store these materials in the reactor canal for a period of time to permit decay of activity to the point where the various pieces can be removed from the canal with a minimum of shielding.

Periodically, this scrap is removed from the canal in large garbage cans, placed in a lightly shielded two-wheel trailer on the end of a long pull bar and hauled to the burial ground for disposal.

In case the activity level requires the use of substantial shielding, the material is transported in a shielded carrier and removed for burial by semiremote techniques.

Engineering test reactor: The engineering test reactor is a high flux, light water-cooled and moderated reactor which can accommodate large experiments in the core and surrounding reflector. The reactor core components, reflector pieces, and experiment containment equipment are cooled by high purity water. There is no air-cooled portion of the reactor as there is in the MTR.

While operation of the reactor produces radioactive water and solids, the experimental program scheduled for conduct in the reactor will yield radioactive liquids, gases, and solids which require collection, monitoring, and disposal under procedures similar to those employed at the MTR.

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Investment is $584,000 for liquid collection and processing, $264,000 for gas collection, monitoring and disposal, and $1,200 for solid waste disposal.

Liquids: The principal source of radioactive liquid waste from the ETR is the primary cooling water stream which is continuously purged with fresh water at the approximate rate of 50 gallons per minute through various gland seals on pumps and other equipment. Water removed from the system is piped to the MTR retention basin where it is stored and disposed of along with liquid waste from the MTR.

In order to maintain the high purity of the primary cooling water and to remove any material added to the water by corrosion or experiment failure, the ETR is equipped with a demineralizer system which can be employed to continuously treat a portion of the circulating primary coolant.

Secondary sources of radioactive water are the warm and hot drain systems which purge water from experiment installations. These drain systems collect waste in separate catch tanks where it is retained until it is analyzed to determine proper disposition.

If the activity is low enough to permit inclusion in the MTR bulk waste, the contents of the catch tanks are pumped to the MTR retention basin. In the event the activity is too high for direct disposal, the water is pumped to the MTR hot storage tanks for treatment with similar waste from the MTR.

Gases: Cooling air from certain experiments and ventilation air from experiment cubicles comprise the waste gases from the ETR which are potentially radioactive. The experiment cooling air which will be radioactive due to activation of argon may at times acquire additional activity due to intentional or accidental leakage in a test specimen.

In order to remove particulate material from the exhaust air, the stream will be filtered prior to discharge to the atmosphere through a 250-foot-high stack.

The experiment cubicle ventilation exhaust, which normally contains no activity, is discharged to the atmosphere through stack. The combined stream is continuously monitored for activity level and total activity discharged.

Under normal operating conditions no problem is anticipated in meeting "Handbook 52" limits with respect to activity release. However, a combination of adverse weather conditions and a leaking experiment could give rise to a temporary situation requiring suspension of experiment operation.

Solids: The types of solid radioactive materials requiring disposal at the ETR will be essentially identical with solid materials requiring disposal at the MTR and will be handled and disposed of in a similar manner.

Representative HOLIFIELD. Thank you, Mr. Doan, for your statement.

Are there any questions of Mr. Doan?

Senator HICKENLOOPER. Yes, I would like to explore a particular field just a minute.

This is a pretty expensive operation, Mr. Doan?

Mr. DOAN. Indeed it is.

Senator HICKENLOOPER. I am not complaining about the safeguards that have to be taken to assure maximum safety in these matters and I may be a little bit in error but I notice on page 3 you say at the top of the page:

> The volume of waste stored varies from 50 to 150 gallons per pound of uranium recovered.

Then we get down to page 4, and you discuss costs per gallon and the average cost runs somewhere between $2.63 a gallon up to $4.90 a gallon, which is apparently the equipment cost.

I was thinking of 150 gallons per pound—and this may be extravagant—150 gallons per pound of uranium and taking the lower figure of $2.63 per gallon it comes to $394.50 per pound of uranium.

I also considered gold at $35 an ounce. Twelve ounces of gold makes it $420 per pound of gold. It gets a little bit expensive.

Mr. DOAN. I believe I emphasized that the $2.63 per gallon represents the cost of storing aluminum waste materials, first cycle waste materials.

The $4.90 represents the cost for zirconium materials. There are some special problems in the processing of zirconium not present in the aluminum.

Senator HICKENLOOPER. I did not take the $4.90. I took the $2.63. I was a little hesitant to multiply $4.90 by 150 gallons.

Now, do these costs that you set out, and that I have referred to by way of illustration, reflect the capital investment in the storage tanks, the cooling system, and all that, or do they include what we call the capital investment in physical equipment and the cost of day-by-day processing, that is, the cost of surveillance, examination, and testing and so on?

Mr. DOAN. The cost per gallon given represents only the investment part of the cost, the investment in tanks and the associated piping to get the materials there.

Senator HICKENLOOPER. So that for want of a better term, you might call that a capital investment?

Mr. DOAN. This is the capital investment cost.

Senator HICKENLOOPER. Now, I believe you said there had been no leakage as yet from these tanks?

Mr. DOAN. That is correct.

Senator HICKENLOOPER. Is it fair to assume that there will come a time when these tanks will actually disintegrate or deteriorate to the point where something has to be done?

Mr. DOAN. I presume there is conceivably such a time and, of course, the fact that we have it set up to be able to detect leaks means that we want to be prepared for that.

The actual life of these tanks, I think, has not yet been established.

Senator HICKENLOOPER. I understand it has not been. And you are equipped so that at the slightest indication of a rupture of one of these tanks, you can be warned and make such transfer as is necessary?

Mr. DOAN. That is correct; yes.

Senator HICKENLOOPER. The only reason I brought this out was that I thought it might be emphasized that this whole business is an expensive proposition.

Mr. DOAN. Indeed it is, although it should be remembered that the cost of storing the waste is still only about five percent of the value of the U–235 recovered.

Senator HICKENLOOPER. I think the whole cost of this operation is vastly greater than the public on the average believes. I think we ought to go into this with our eyes open and know what we are up against and the various collateral and direct or indirect costs of our activities in the atomic energy field are really very, very substantial.

Representative HOLIFIELD. Do you not believe, Senator, that the honest economics on the cost of power must include all of these corollary costs?

Senator HICKENLOOPER. I not only believe that, but for some time I have been trying to insist that the cost per kilowatt hour at buss bar, let us say, that we have been talking about, is not a realistic figure as to the actual cost of that kilowatt.

While I want to encourage the development of atomic electric power, I do not think we ought to be Pollyannish about it. I think we ought to look at the thing squarely in the face and realize that it is substantially more costly in total dollars regardless of who puts it in, it is vastly more costly than a great many people believe.

Mr. DOAN. Senator, I would like to be sure you understand when we talk about this cost per pound of uranium recovered, it represents uranium 235 and not natural uranium.

Senator HICKENLOOPER. As I said, I do not mean to be facetious about this. As I said at the outset perhaps I took an extravagant illustration to emphasize the point.

The point I used is probably not applicable all the way down the line. But the point remains that there are a lot of concealed costs in my opinion, in this matter, that are not permitted to be reflected in much of the discussion that goes on in the development of atomic energy.

Mr. DOAN. I have the impression that they really are reflected. Those who are figuring the costs of nuclear power certainly have this information available and I feel the total processing cost that is given does indeed reflect the storage costs as well as the actual chemical processing cost.

The fact that the cost is high is the chief impetus behind the research going on as Dr. Morgan indicated at Oak Ridge and elsewhere, including our own place.

Senator HICKENLOOPER. I understand, and I believe these figures are all available, but it depends on how many sets of books you keep.

They tell me in some countries for tax purposes a man may keep three sets of books. One that he keeps for his partner, one for himself, and one for the Government. It depends on what they charge a lot of these costs to, which column they put them in on the accounting sheet.

I think, too often, many charges that legitimately ought to go into a realistic cost structure may be for one reason or another charged to another item.

But there is no use getting into that kind of discussion here. I just wanted to emphasize the high cost of it.

Senator PASTORE. Touching upon the point that has been developed by Senator Hickenlooper, I was wondering if you could say it in more mundane terms, in a simplified fashion that a layman can understand, what part of the overall kilowattage cost would this responsibility of waste storage constitute? Can you say that in a fashion that the man on the street can understand?

Mr. DOAN. I do not have the exact figures on this, but it is a relatively small fraction of the overall nuclear power cost. We have had no occasion, ourselves, to figure the overall economy of particular reactors. Certainly the fuel cost has to include reprocessing and waste storage.

If we could find cheaper methods of processing and storing, we would to that extent cut down on the overall cost of power.

Senator PASTORE. Let me ask you a general question along that line: From your knowledge of it, would you consider it a discouraging factor on the development of atomic energy for electric power?

Mr. DOAN. No, at the present time I would not consider it a discouraging factor.

There are many factors we have to explore further. There are research programs going on that will lead to cheaper ways of storing as well as cheaper ways of processing.

Senator PASTORE. I have been told by Mr. Toll that that will be developed next Monday.

Now, you are a man of private industry. Do you see this storage as completely a private enterprise and private responsibility or do you envision this as a public responsibility because of the indefiniteness of this surveillance that is mandatory in this sort of thing? Do you understand my question?

Mr. DOAN. Yes. I am sure it has to be both. I believe it has to be a private responsibility in the sense that if private money is invested in building and operating the reactors, a part of that responsibility certainly has to be the storage problem.

I think, however, that we could all agree that the storage of radioactive waste materials must always be a public concern and be subjected to necessary regulations to protect the health of the public.

Senator PASTORE. We are talking here about a thousand years, which, of course, is a long time.

How can you invoke private responsibility and a responsibility that might span over a thousand years?

Mr. DOAN. That is a difficult question. I think that all we can do at any one time is to consider this a responsibility of private industry, The very long time required to ultimately reduce the radioactivity to a stage where it will not present any public hazard will necessitate setting up the regulatory and inspection function on a long-term basis.

Senator PASTORE. I hope you do not misunderstand me, Doctor, because I am one on this dais here who feels we should not be allowed to be discouraged in the least, nor should we relax our efforts to promote the research and development of atomic energy as far as we can go and as far as we can go for the exploitation of the atom, itself.

But now that you have presented this picture of indefiniteness that you have been talking about and the fact that the public will be interested in this whole field of waste disposal and storage because of the time element involved and because of the safety and the health aspects involved, I am wondering how we are going to knit together and weld together the private aspect of the development of electric power for atomic energy and weld into it, too, the comparable responsibility of the public if the public, the chances are, will not be interested in the profits that will ensue from the development of the peaceful atom.

Mr. DOAN. It certainly is a complicated question for which the policy, like a good many other things in atomic energy, will have to be developed as times goes on. However, I am firmly convinced that the public interest will best be served by placing the responsibility for safe and economical storage of radioactive waste materials squarely in the shoulders of private industry with appropriate Government regulation.

Senator PASTORE. You see, there are many people who feel there has been a hesitancy on the part of private industry because government has remained to a large extent in the surveillance, in the direction and in the investment and in the responsibility of the peaceful development of atomic energy.

It has been resisted and resented by some people who feel they have not moved quite as fast as they might have if you did not have so much governmental surveillance.

Yet as we develop the various aspects and the various facets of this whole field, we realize how much of it must remain in the public domain for the obvious reasons, as we have said, of the time element in the disposal of waste, and the surveillance necessary and health and safety aspects involved.

The question I want to pinpoint with you today is if in your fair judgment, taking this field of waste alone, and storage, you envision here governmental supervision in one form or another ad infinitum.

Do you understand that question?

Mr. DOAN. Yes, I am sure there must be some sort of Government surveillance.

Now, I do not mean that it has to be a direct sort of thing. It should preferably be through suitable regulation rather than direct supervision as we have it now.

This is always going to be a matter of public concern, the continuing storage problem of radioactive waste. We are not going to be able to get away from that.

Senator PASTORE. Do you envision at any time that laws will be drawn directing the public utilities to find their own burial ground, to make its own storage facilities and maintain them in the public interest and for public protection and safety, or do you envision that at some place in the United States or some place in the world, there will be public burial grounds under governmental supervision or control whereby a certain charge will be leveled or directed against the public utility company, but in essence this is something that has to be watched very carefully, cannot be entrusted to other people who may be interested in profits and loss and will have to be directed exclusively by government or can be delegated to private industry.

What is your objection to that?

Mr. DOAN. I see nothing basically in the problem of disposal of solid radioactive waste that could not be handled by some sort of private industrial enterprise. The activities of such enterprises would have to be subject to close governmental regulation as in a good many other cases of industrial operations presenting certain problems and hazards to the public.

Chairman DURHAM. Will the Senator yield at that point?

Senator PASTORE. Yes.

Chairman DURHAM. I think the question you asked about this problem is already facing us because some States have already adopted laws, with very little knowledge, in my opinion, as to the problem of control of radiation waste material.

That is one thing that concerns me very deeply.

Senator PASTORE. At the present time I daresay that much of the problem has a very great public connection for the reason that there has not been too much private development.

I am projecting myself 50 or 100 years from now when these atomic plants will be rather common all over the country, where the waste problem will become greater and greater as time evolves and how we begin planning.

I understand we are in the experimental stage now and you are hoping to learn how to handle this waste in better fashion and more economically.

I think, too, we ought to have a proper relationship between waste disposal and the cost per kilowatt of electric power to the development of atomic energy.

But the fact still remains that as time goes on we have to give considerable thought to the safety aspects involved and to the cost aspects involved.

You say now as you look at it time will tell, but as of the moment you are quite optimistic that this problem can be worked out without any discouragement to the program of accelerating research and development for peaceful uses of atomic energy.

Mr. DOAN. I would say so, yes.

Chairman DURHAM. Doctor, your record is excellent of course, in all your operations, we are aware of that, in the operations of all plants in all sections of the country.

This question should probably be directed to Mr. Horan:

Have you had any accidents or exposures from radioactive materials so far in your operations.

Mr. HORAN. Nothing of any significance. There have been minor releases of activity. This is basically airborne activity.

Chairman DURHAM. But nothing of a dangerous nature?

Mr. HORAN. No. There has been no exposure in excess of 10 percent of the amount that is permitted to the offsite population.

Chairman DURHAM. Which is safe?

Mr. HORAN. That is correct.

Representative HOSMER. Along the line of the Senator's inquiry, there has been a considerable body of cemetery law developed in this country, and a law in relation to perpetual care funds and so forth.

I wonder if there would be any analogy to the waste disposal problem that could be drawn from that body of law?

Mr. DOAN. I must say I am not sufficiently familiar with that to offer an opinion.

Representative HOSMER. Your organization has not looked into that?

Mr. DOAN. I believe not.

Representative HOSMER. Thank you.

Representative HOLIFIELD. Are there any further questions?

Thank you, Mr. Doan, for your testimony. We will insert your formal statement in the record.

(The statement referred to follows:)

STATEMENT OF PHILLIPS PETROLEUM CO. AS OPERATOR OF THESE FACILITIES UNDER CONTRACT WITH THE ATOMIC ENERGY COMMISSION RE WASTE MANAGEMENT OPERATIONS PROGRAM ON THE COLLECTION, HANDLING, PROCESSING AND DISPOSAL OF WASTE MATERIALS AT THE IDAHO CHEMICAL PROCESSING PLANT, MATERIALS TESTING REACTOR AND ENGINEERING TEST REACTOR

The Materials Testing Reactor, Engineering Test Reactor and Idaho Chemical Processing Plant operated for the Atomic Energy Commission by Phillips Petroleum Co. all generate radioactive liquid, gaseous and solid waste materials which must be stored or disposed of in such a manner as not to constitute a current or future hazard to people working at the National Reactor Testing Station or to people, animals and property in areas surrounding the site. To this end, every reasonable precaution is exercised to insure that materials which must be retained are stored safely and that materials which may be released in prescribed quantities to the atmosphere or to the earth are released under carefully monitored and controlled conditions.

In the case of liquids discharged from the plants and returned to the earth via leaching ponds or disposal wells, the average concentration of radioactive material in the liquids is kept within limitations prescribed by the AEC, which are based on information contained in National Bureau of Standards Handbook 52. The same is true of waste gases discharged to the air. Constant monitoring outside the perimeter fences of the plants, conducted by the Site Survey Branch of the Idaho Operations Office Division of Health and Safety, insures that the rates of radioactive gaseous waste discharge together with prevailing meteorological conditions do not give rise to bulk air contamination which transcends established limits.

Radioactive solid waste materials are not subject to discharge rate control. These materials are buried in a burial ground operated and controlled by the Health and Safety Division of the Idaho Operations Office. Solid materials which are insoluble in water are buried directly in the ground, while soluble materials are suitably packaged in waterproof containers before burial.

Since the three plants under the jurisdiction of Phillips Petroleum Co. do not have identical problems of waste handling, each unit will be treated separately, starting with the Idaho chemical processing plant which generates the largest quantity of radioactive waste materials.

IDAHO CHEMICAL PROCESSING PLANT

The ICPP is devoted principally to the recovery of enriched uranium from spent reactor fuel elements. The recovery process involves dissolution of the fuel elements in acid followed by extraction of essentially all of the enriched uranium from the resulting solutions. The reprocessing operation generates a liquid waste containing most of the nonvolatile fission products produced during reactor operation and a gaseous waste containing radioactive krypton and iodine.

Liquids

The principal sources of radioactive liquid wastes are the three cycles of liquid-liquid extraction by which the enriched uranium is recovered from the dissolved fuel elements. The aqueous waste material from the first step contains the greatest quantity of fission products while the waste from the third cycle contains the least. Because of the quantities of contained fission products, all of these wastes are stored for an indefinite period in 300,000 gallon underground stainless steel tanks. Due to the quantities of heat-generating fission products in the first cycle waste, it is necessary to store this material for several years in water-cooled tanks to eliminate boiling and reduce corrosion of tank materials. The second and third cycle wastes are stored together in uncooled tanks. The volume of waste stored varies from 50 to 150 gallons per pound of uranium recovered, depending upon the uranium content of the fuel being processed.

As the waste material flows from the extraction columns, it is temporarily collected in small storage vessels to permit analysis for residual uranium. If the uranium content is satisfactorily low, the waste is fed to evaporators where part of the water is removed to effect volume reduction with resultant economy in storage space. From the evaporators the waste is transferred directly to the appropriate underground storage tanks. The tanks are so installed that a leak can be detected immediately. In the event of a leak the contents of the tank would be transferred to an empty tank which is always available to meet such an emergency. There have been no leaks to date.

Secondary sources of radioactive liquid wastes are the process equipment cell floor drains, some laboratory drains, equipment decontamination solutions and the water removed from the primary process wastes during partial evaporation. These wastes are normally low in activity but above levels which can be returned directly to the area water table. Waste materials from these sources are collected in small underground tanks, sampled for uranium content, and then fed to an evaporator where most of the water is taken overhead at an activity level low enough to permit disposal, after reasonable dilution, directly to the area water table by means of a disposal well. The bulk of the fission products which enter the evaporator with the feed stream remain in the kettle product and are transferred to permanent storage in the large underground tanks.

All liquid discarded in the disposal well is continuously sampled in order to provide a daily record of total radioactivity placed in the water table. In addition, instruments constantly monitor the stream to the well and sound an alarm in the event the activity reaches a preset level. The average level of activity discharged to the water table is held within the limits established by the Idaho Operations Office of the Atomic Energy Commission.

The collection, processing, monitoring and storage of liquid wastes are costly items, both from an investment and operating viewpoint. The collection of these wastes, including the necessary piping, intermediate storage tanks, sampling systems, transfer jets, instrumentation and housing, requires an investment of $3,830,000 at the ICPP. The evaporators, piping, sampling system, instrumentation and housing incidental to processing the liquid wastes for storage represent another $4,955,000 investment. Cost of monitoring equipment is $155,000. Finally, the total investment in storage tanks is $7,680,000 made up as follows: nine 300,000 gallon permanent storage tanks for aluminum waste of which six are cooled. Average cost is $789,000, which is equivalent to a storage cost of $2.63 per gallon including apportioned piping. Four 30,000 gallon tanks are provided for storage of zirconium waste. Total cost was $580,000, which is equivalent to a per gallon cost of $4.90.

Gases

During dissolution of the fuel elements, some gaseous radioactive fission products are released inside the dissolver vessels. These gases are removed under vacuum through a dissolver off-gas system which retains some of them, and the residue is filtered to take out radioactive particulate matter, diluted with air and then discharged to the atmosphere through a 250-foot high stack. For normal plant operation the dilution afforded by plant ventilation air, plus the additional dilution effected by dispersal to the atmosphere from the top of the stack, yields an activity level at the point of exposure which meets established limits.

In a separate process for the recovery of radioactive barium, large quantities of radioactive iodine and xenon are contained in the evolved waste gases. The quantities and release rates of these materials are such that except under very favorable meteorological conditions it is not permissible to release them directly to the stack. In order to stay within prescribed limits the gases containing iodine and xenon are scrubbed with caustic solution to partially absorb iodine, passed through activated charcoal to further reduce the iodine content, and, if required by meteorological conditions, can be partially retained in a 10,000 cubic foot gas holder for gradual release to the atmosphere under favorable weather conditions. During these operations which are conducted on an average of 1 day each month, constant liaison by radio exists between operations personnel responsible for the process and representatives of the U.S. Weather Bureau and the Health and Safety Division of the Idaho Operations Office to insure that the quantities of iodine and xenon released do not constitute a hazard within or outside the National Reactor Testing Station.

The total investment in off-gas collection equipment at the ICCP is $1,235,000, which includes the vessel off-gas system, the dissolver off-gas system and incidental instrumentation. There is also a $125,000 filter system, and an investment of $250,000 for the stack, stack heater, and duct work, with an additional $70,000 allocated for improvement in stack gas monitoring.

Solids

Minor amounts of radioactive solid waste materials require disposal from the ICPP. These materials which comprise worn process equipment, rags and mops used in decontaminating equipment, and other small items are appropriately packaged and hauled to a burial ground for permanent disposal. De-

contamination and disposal equipment for handling these solid wastes is valued at $30,000.

Approximately 50 employees at the ICPP are engaged full time in handling various phases of the collection, processing, monitoring, and disposal of liquid, gaseous, and solid radioactive wastes at the ICPP. The total annual operating and maintenance costs are $919,580, made up of $666,180 direct and $253,700 indirect expense.

MATERIALS TESTING REACTOR

The materials testing reactor is a high-flux light water cooled and moderated reactor designed to accommodate a large variety of experiments ranging from small capsules to large high-temperature high-pressure equipment rivaling in complexity the reactor itself. The fuel elements, control rods, and other equipment surrounding the reactor core are cooled by high purity water, as are many experiments in and surrounding the core. A graphite cube and heavy steel thermal shield which surround the reactor tank are cooled by air which is drawn through the graphite from the reactor building and subsequently discharged to the atmosphere.

Operation of the reactor and the various experiments gives rise to liquid, gaseous, and solid wastes which are radioactive and must, therefore, be disposed of in a manner consistent with established health physics requirements. The air and water which cool the reactor become radioactive due to neutron bombardment. Various experiments may from time to time give rise to liquid and gaseous waste which must be disposed of along with the principal waste from the reactor. Solid wastes are principally sections of experimental equipment and portions of fuel elements and control rods which become radioactive while in the neutron atmosphere of the reactor.

Investment is $610,000 for liquid collection, processing, monitoring, and disposal; $342,000 for gas collection, monitoring, and disposal; and $3,700 for solid disposal. Annual operating and maintenance costs for MTR and ETR are $78,000 direct costs and $30,400 indirect.

Liquids

The bulk of radioactive liquid waste generated at the MTR emanates from the primary process water system used to cool the reactor core components. While this water is demineralized to high purity prior to use, it still contains residual traces of minerals, principally sodium 23, which upon passing through the core of the reactor become radioactive. In addition, small quantities of other elements are picked up by the water from the piping, the reactor tank and core components, and experiments installed in and around the core; these elements also become radioactive and add to the total activity in the water system. Occasionally, an experiment which is being cooled by the process water may rupture or otherwise leak with a resultant release of fission products or activated elements directly to the cooling water. Thus, we have three sources of activity in the primary cooling system with activated sodium normally contributing the major portion of activity.

In order to prevent the gradual build up of trace elements in the primary cooling water, a portion of the stream is constantly bled off and replaced with fresh water. The water removed from the system is fed to one of two 350,000-gallon concrete retention basins where the average time required to flow from inlet to outlet is equivalent in decay time to 10 half lives of the radioactive sodium. By the time the water has spent 150 hours in the retention basin the activity has reduced to the point where the water may be returned to the ground by means of a leaching pond. On occasions when the activity in the process water stream has been raised by a leaking experiment to the point where normal disposal cannot be effected, a bypass demineralizer is employed to continuously demineralize a portion of the primary water until the total activity is reduced to a level which can be safely handled by normal disposal procedures. The demineralizer resin is capable of removing from the water most fission products and source materials which can be added to the water from experiment failures. When the resin reaches its capacity for adsorption of these materials, it is removed in sealed steel tanks and buried with other solid wastes.

A secondary source of radioactive liquid waste is a system which collects "warm" and "hot" wastes from laboratory drains, experiment installations, and floor drains on and around the reactor. This system collects the waste in small catch tanks where it is held until the major contaminants are determined and a measurement made of the activity level. If the activity level and contaminant

content are such that the waste can be discharged to the water table, it is pumped to the previously mentioned retention basin for disposal along with the purge stream from the reactor. In he event the activity level of the waste in a catch tank is too high for direct disposal, it is transferred to one of two 10,000-gallon storage tanks to await treatment for removal of radioactive materials by ion exchange in one of four demineralizers at the engineering test reactor. From the demineralizers the cleaned waste is returned to ground via the retention basin and leaching pond.

Gases

The cooling air which removes heat from the thermal shield and graphite in the MTR becomes radioactive as it passes through the reactor due to activation of argon which is a minor constituent of air. This air flows through the reactor at the rate of 24,000 cubic feet per minute and is discharged to the atmosphere through a 250-foot high stack. The effluent from the stack is constantly monitored to yield information on activity level and total activity discharged. In addition to the radioactive air from the reactor cooling system, there are contributions to the total made by the reactor cooling water degassing system, certain experiments and laboratory hood exhausts. These streams are combined with the reactor cooling air exhaust and discharged through the stack. In the event of experiment rupture in the reactor tank, the activity contributed by the cooling water degassing system may exceed that from argon activation in the cooling air. However, due to the use of a bypass demineralizer on the cooling water stream, this contribution is short lived and to date has not created a disposal problem.

Solids

Radioactive solid materials which require disposal are (1) fuel element and control rod end boxes, which are removed prior to reprocessing to save cost; and (2) parts of experimental equipment, which has served its purpose but has become radioactive due to exposure in the reactor. These materials normally would be looked upon as scrap in a non-atomic-energy industry, but because of the induced activity cannot be handled as scrap through industrial channels.

It is normal practice to store these materials in the reactor canal for a period of time to permit decay of activity to the point where the various pieces can be removed from the canal with a minimum of shielding. Periodically, this scrap is removed from the canal in large garbage cans, placed in a lightly shielded two-wheel trailer on the end of a long pull bar and hauled to the burial ground for disposal.

In case the activity level requires the use of substantial shielding, the material is transported in a shielded carrier and removed for burial by semiremote techniques.

ENGINEERING TEST REACTOR

The engineering test reactor is a high-flux light water cooled and moderated reactor which can accommodate large experiments in the core and surrounding reflector. The reactor core components, reflector pieces and experiment containment equipment are cooled by high purity water. There is no air-cooled portion of the reactor as there is in the MTR.

While operation of the reactor produces radioactive water and solids, the experimental program scheduled for conduct in the reactor will yield radioactive liquids, gases, and solids which require collections, monitoring, and disposal under procedures similar to those employed at the MTR.

Investment is $584,000 for liquid collection and processing; $264,000 for gas collection, monitoring, and disposal, and $1,200 for solid waste disposal.

Liquids

The principal source of radioactive liquid waste from the ETR is the primary cooling water stream which is continuously purged with fresh water at the approximate rate of 50 gallons per minute through various gland seals on pumps and other equipment. Water removed from the system is piped to the MTR retention basin where it is stored and disposed of along with liquid waste from the MTR. In order to maintain the high purity of the primary cooling water and to remove any material added to the water by corrosion or experiment failure, the ETR is equipped with a demineralizer system which can be employed to continuously treat a portion of the circulating primary coolant.

Secondary sources of radioactive water are the "warm" and "hot" drain systems which collect purge water from experiment installations. These drain systems collect waste in separate catch tanks where it is retained until it is analyzed to determine proper disposition. If the activity is low enough to permit inclusion in the MTR bulk waste, the contents of the catch tanks are pumped to the MTR retention basin. In the event the activity is too high for direct disposal, the water is pumped to the MTR "hot" storage tanks for treatment with similar waste from the MTR.

Gases

Cooling air from certain experiments and ventilation air from experiment cubicles comprise the waste gases from the ETR which are potentially radioactive. The experiment cooling air which will be radioactive due to activation of argon may at times acquire additional activity due to intentional or accidental leakage in a test specimen. In order to remove particulate material from the exhaust air, the stream will be filtered prior to discharge to the atmosphere through a 250-foot-high stack. The experiment cubical ventilation exhaust, which normally contains no activity, is discharged to the atmosphere through the stack. The combined stream is continuously monitored for activity level and total activity discharged. Under normal operating conditions no problem is anticipated in meeting Handbook 52 limits with respect to activity release. However, a combination of adverse weather conditions and a leaking experiment could give rise to a temporary situation requiring suspension of experiment operation.

Solids

The types of solid radioactive materials requiring disposal at the ETR will be essentially identical with solid materials requiring disposal at the MTR and will be handled and disposed of in a similar manner.

Representative HOLIFIELD. Mr. A. L. Bethel of Westinghouse, who is in charge of waste management operations at Shippingport, will be the next witness.

STATEMENT OF A. L. BETHEL,[4] BETTIS ATOMIC DIVISION, WESTINGHOUSE ELECTRIC CORP.

Mr. BETHEL. Mr. Chairman, I have submitted for the record a detailed technical report on the waste management operations at Shippingport, and would like to give you a fairly brief nontechnical summary of those operations.

Representative HOLIFIELD. We would like to have that. Of course, you know we are running behind, so will you move right along.

Mr. BETHEL. Yes, sir. The location of our country's first full-scale nuclear power station at Shippingport, Pa., has made the waste dis-

[4] Born: Joplin, Mo., Oct. 6, 1921.

Attended Joplin Junior College, 1939–41 in chemical engineering. Entered U.S. Military Academy, West Point, N. Y., in July 1941. Graduated in June 1944 and commissioned second lieutenant, U.S. Army Corps of Engineers. Served in various field assignments, including assistant manager of operations, manager of operations, Manila, Engineer District, Manila, P.I., in 1946. In the fall of 1946, returned to United States to an assignment with the Manhattan project. Worked in the atomic weapons program at Los Alamos and Sandia Base, Albuquerque, N. Mex., until 1948. Was selected by the Army for postgraduate educational program. Studied at U.S. Naval Post Graduate School, Annapolis, Md., at the Johns Hopkins University, Baltimore, Md. Received masters degree in physics from Johns Hopkins in June of 1951. From June of 1951 to April of 1955 was assigned to Headquarters, Armed Forces special weapons project, Washington, D.C., as special assistant to the Chief of the project.

Resigned from the Army in April 1955 and joined Westinghouse at Bettis plant. Has worked continuously on the PWR project, joining Westinghouse holding the following positions: Assistant to powerplant manager; assistant to the project manager; manager, PWR test and operations group; manager, PWR Shippingport activity. Presently, manager, PWR powerplant engineering.

Mr. Bethel had responsibility for Westinghouse for the PWR checkout and test and preliminary operational phase up to synchronization of the generator with the Duquesne Light Co. system.

posal facilities there a part of a successful operation. The station is in a populated area and it is also on the Ohio River. The design personnel consulted with and obtained advice from the U.S. Public Health Service, the Pennsylvania department of health, and the Ohio River Valley sanitary authority, in addition to the Atomic Energy Commission and some of its other prime contractors.

Maximum permissible concentrations for radioactive pollutants have been established by the National Committee on Radiation Protection, and the design objective was to assure that these tolerance values would not be exceeded.

The design basis of the PWR waste disposal facilities was that, when necessary, all radioactive wastes must be reduced in concentration exclusive of natural background and atomic weapons testing such that the concentration in water measured at the effluent water stream before discharge to the Ohio River and in air at the point of discharge to the atmosphere shall not exceed one-tenth the maximum permissible concentration—MPC—as recommended in the National Bureau of Standards Handbook Nos. 61 and 52.

There are fundamentally three sources of radioactivity in PWR. First, of course, are the fission products.

As you may recall, the fuel in PWR is zircaloy clad and this cladding prevents release of the fission products from the fuel to the circulating water. However, some of the fuel elements may develop holes which would allow some of the fission products to escape to the coolant. The plant and waste disposal facilities have been conservatively designed to allow operation with up to 1 percent, or approximately 1,000, defected fuel elements. The Shippingport facilities will not be required to handle all of the fission products because, when the core is depleted, it will be shipped to another facility for reprocessing.

The second source of radioactivity in PWR is the activation of impurities present in the reactor coolant. Although all surfaces in contact with the coolant are constructed of highly corrosion-resistant materials—stainless steels and zircaloy—some corrosion does occur. The corrosion reaction produces soluble metal oxides and very fine particles of insoluble materials which become activated by neutrons in passing through the reactor core.

The last source of activity is the formation of tritium. In order to keep corrosion of the primary system piping and components at a minimum, lithium hydroxide is added to make the water slightly alkaline because this condition greatly enhances the corrosion resistance of the stainless steel. In passing through the reactor, a small fraction of the lithium interacts with neutrons and forms tritium. The formation of tritium is preferable to the formation of radioactive corrosion products because of the comparatively small biological hazards associated with tritium.

The classification of radioactivity by source was used to estimate the quantity and activity of isotopes present in the primary coolant water. The designers studied the various methods by which the wastes would be discharged from the reactor plant, because the form in which the wastes occur dictates to a large extent the processes which may be used for efficient disposal.

I have some charts (see PWR presentation figs. 1 and 2, pp. 656 and 657) which I will use to discuss the form in which the disposal plant receives wastes and the processing done before release.

By far the largest source of activity is the reactor plant effluent, which is radioactive waste liquid composed primarily of discharged reactor coolant water. These liquids, containing both volatile and nonvolatile radioisotopes, are received and stored in large underground surge tanks. During the storage period, normally a maximum of 45 days the activity is reduced by natural decay.

Following this, if necessary, the liquid is processed through purifiers called demineralizers or ion exchangers. A reduction of about 1,000-fold in the nonvolatile activity is obtained as the radioisotopes are collected on the resin.

The water is next passed through a gas stripping column where essentially all of the gas is removed. The water is then held in a test tank for final sampling prior to discharge to the waste effluent channel. The rate of discharge is determined from the results of sampling so that the wastes when mixed with the turbine plane condenser cooling water will produce a final effluent that contains no more than one-tenth of the MPC of radioactivity for drinking water.

The discharge of waste liquids is continuously monitored, and an alarm is provided to indicate an approach to overtolerance conditions.

Liquid wastes which have a high content of dissolved solids are made up principally of solutions that are used for decontaminating reactor plant components. The high amount of dissolved solids in these wastes precludes the efficient use of the ion exchange process. The liquid is therefore evaporated to concentrate the radioactivity, and the concentrated liquid is mixed with cement and drummed for ocean burial.

The distillate is discharged to a surge tank and is treated from that point on as reactor plant effluent.

Monitored wastes include the effluents from the various laboratory drains, personnnel showers, and the hot and cold laundries. These wastes, normally only slightly radioactive, are collected and sampled in separate tanks. When sampling verifies that the activity is low enough, they are discharged at a controlled rate and diluted in the condenser cooling water stream so that the final discharge to the river is less than one-tenth MPC. Should monitored wastes be of such activity that discharge by dilution would be too time consuming, they are processed through the evaporator.

The reactor coolant contains trace amounts of radioactive gases, primarily xenon 133 and krypton 85. All tanks and vessels which contain reactor plant effluent are continuously flushed with a recirculating gas stream. The gases from the stripping column are also discharged into the flush gas system. At infrequent intervals gas is bled from the system into drums for storage and decay of the radioactive xenon and krypton. After a sufficient decay period, which may be as long as 60 days, this gas is mixed with the dicharge of an air dilution fan and released to the atmosphere. This final mixture is continuously monitored and an alarm is provided to indicate approach to overtolerance conditions.

The other waste disposal processes used at Shippingport are shown on the next chart.

Resin from both the reactor plant purification system demineralizers and the waste disposal demineralizers is slurried and pumped to one of the two underground stainless steel storage tanks. These tanks were designed to provide storage capacity for a minimum of 10 years. As you can see on the chart the tank in turn is enclosed in concrete waterproof enclosures.

An incinerator is provided to reduce the volume of radioactive combustible solid wastes. Particulate matter in the flue gas is removed by a wet gas scrubber and a high efficiency filter prior to being mixed with the discharge of the air dilution fan and release to the atmosphere.

In the original design the residual ash was mixed with water and pumped to an underground storage tank. This proved to be impractical because nails, nuts, bolts, and the like, in the ash, would foul the transfer pumps, so we have gone to a procedure for direct drumming of the ash for ocean burial.

Small noncombustible contaminated solid wastes such as small parts, tools, et cetera, are imbedded in concrete and drummed for ocean burial. Large items may be buried at the site, although nothing has received this disposition to date.

The processes which have been described are essentially batch-type. Efficient control is obtained by sampling after each individual processing step. Any batch may be reprocessed, if necessary, prior to discharge to the environment. Only one liquid or gas stream is discharged at a time for maximum control and safety.

The predicted average monthly accumulation of reactor plant effluents was 23,000 gallons. In the first 5 months of operation, up to 14 times this design quantity was discharged per month.

Special operating conditions necessitated by the test program carried out during the period, accounted for most of the excess. Since the activity level during this period was less than one ten-thousandth of the design value, the excess volume was not a problem. As the mode of plant operation has approached that of a typical central power station, the quantity of reactor plant effluents has decreased to near the design figure, although the activity stayed quite low.

Considering altogether wastes from the hot and cold laundry, personnel showers, and all laboratory drains, the average monthly volume of the monitored waste liquids encountered thus far has been only 30 to 40 percent of design. The combined activity levels have varied between 1 and 10 percent of the design values. Both volume and activity are expected to increase as the degree of plant contamination increases.

The average accumulation of radioactive gaseous wastes was expected to be about 430 cubic feet per month. To date an average of 300 cubic feet per month of gas has been discharged, but the radioactive content has been only about 1 percent of design. The total amount of gaseous activity discharged in the first 11 months is less than 10 percent of the design discharge for a single month.

The amount of combustible solid wastes disposed of in the incinerator has approximated the predicted average of 4,000 pounds per month.

In general, all disposable combustible materials used in an area of possible contamination have been burned. The average activity level per incinerator charge has been near the lower limit of measurability.

The criteria for the discharge of radioactive wastes from PWR are contained in the operating permit granted to the Duquesne Light Co. by the Commonwealth of Pennsylvania. Under the terms of the permit a maximum of about 0.05 curies of radioactivity may be discharged to the Ohio River during a single month. Because of its comparatively high tolerance level, tritium activity is determined separately and has an individual discharge limit of 300 curies per month. The total activity discharged in the first year of plant operation was about 0.04 curies of mixed isotopes and about 50 curies of tritium. Both of these quantities are much less than the permissible discharge for a single month. This is an indication of the margin available for safe disposal of liquids at the higher activity levels which might occur later in plant operation.

For nearly 2 years prior to operation of PWR an environmental radiation monitoring survey was conducted in the vicinity of the Shippingport plant by the Bettis Industrial Hygiene Department. This program is being continued for the Atomic Energy Commission during plant operation by the Duquesne Light Co.

In addition, environmental monitoring is being conducted in the vicinity of the Shippingport plant by the Pennsylvania State Health Department and the U.S. Public Health Service. Periodically, samples of Ohio River water, soil, well water, vegetation, and air in the vicinity are analyzed.

The preoperational surveys provided data on the normal or background radiation levels in the area. Since the reactor went critical in December of 1957, the results of environmental radiation surveys have shown no indication of an increase in area radioactivity levels that could be attributed to operation of the Shippingport plant. Continued comparison of activity in the Ohio River upstream and downstream of the plant shows no detectable increase in background radiation. Such a result is to be expected, based on the known amounts of radioactivity actually discharged to the environment from the waste disposal system.

The Shippingport plant operated at conditions most typical of a central power station during the period July through October 1958. During this period the plant was first operated for 1,000 hours at full power, followed by a period of peakload operation wherein the station supplied power as demanded by the Duquesne Light Co. system. The 4-month period was concluded with a second 1,000-hour run at full power. During the period the net generation of electrical power was over 120 million kilowatt-hours at a plant load factor of 69 percent.

The costs associated with the operation of waste disposal as reflected on an Atomic Energy Commission combined operating report for the period July through October, were as follows: Labor, $20,669; supplies, $6,171; maintenance, $10,360. Total 4 month operating cost, $37,200, an average cost of $9.300 per month.

The continued operation of Shippingport will determine whether these costs are typical and will provide data to establish modified

criteria for the design of disposal facilities for other nuclear power plants in the future.

Representative DURHAM. What is the total cost on a yearly basis?

Mr. BETHEL. The costs cited were extracted from a statement for a 4-month period of typical operation. On this basis the yearly cost would be slightly over $110,000.

Representative DURHAM. What is the capital cost?

Mr. BETHEL. The capital cost of the waste disposal plant was about $2,700,000.

Chairman DURHAM. I want to congratulate you on taking a good deal of time and care because probably you have something which will serve as a model yet, throughout the country. I think you have done an excellent job. You have had no accidents, no radiation exposures in waste disposal operations at Shippingport?

Mr. BETHEL. No, sir.

Representative DURHAM (presiding). Are there any further questions?

Representative BATES. He has already answered the question I had in mind.

Representative DURHAM. Thank you very much, Mr. Bethel. We will include the PWR formal statement at this point, together with other papers on this subject. We will now adjourn until 2 o'clock this afternoon.

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PRESENTATION ON SHIPPINGPORT ATOMIC POWER STATION (PWR) WASTE DISPOSAL FACILITIES

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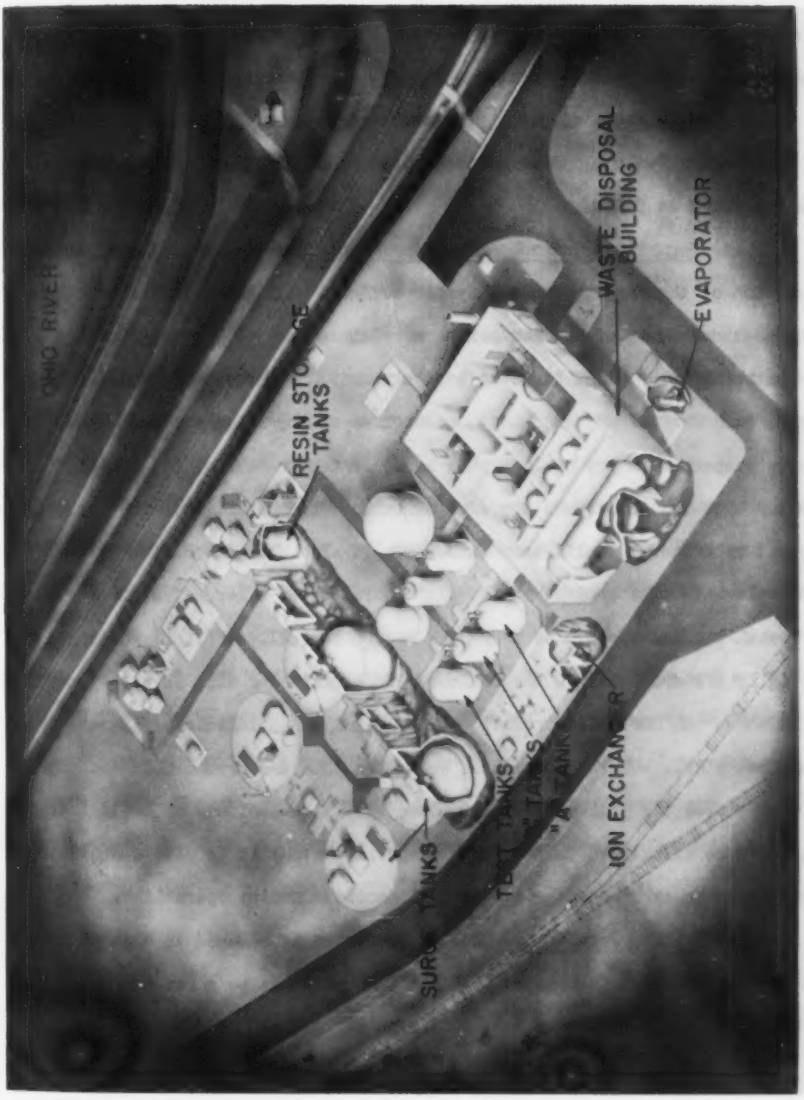
INTRODUCTION

Special waste disposal facilities have been designed and installed for the control of the radioactive wastes from the nation's first full-scale nuclear power station at Shippingport, Pennsylvania. A considerable amount of engineering effort was directed toward the development of the disposal criteria, the waste treatment processes, and the controls required for satisfactory waste disposal at this plant. The major results of this study along with a description of the waste treatment processes are presented in this paper.

The PWR facilities were designed on a conservative basis to provide flexibility for handling waste materials from reactor cores not yet designed or even envisioned. With an objective of reducing reactor operating costs, future cores and future operating procedures may be such as to require the ability to handle greater amounts of radioactivity in the plant. The PWR waste disposal system has therefore been designed to handle far greater radioactivity than has been encountered in the plant to date or than is expected. This will permit, if it should prove desirable, operation of the plant with low quality fuel elements or under various casualty conditions.

As a result of the design and operation of the PWR waste disposal system, it is believed that a better understanding will be obtained regarding the requirements for handling wastes from future pressurized water reactor plants. As yet there is not sufficient basis to determine the cost of a waste disposal system for future pressurized water reactors, nor to state the required manpower and operating costs, since the radioactivity of the PWR wastes has so far been trivial and no special requirements have been imposed on the waste disposal system. A summary of the operations required with associated costs and manpower to date is presented in this report.



CUTAWAY OF THE PWR WASTE DISPOSAL AREA

THE REACTOR PLANT - SOURCE OF RADIOACTIVE WASTES

The source of energy from which power is produced at the Shippingport Plant (PWR) is the heat generated by nuclear fission in the reactor core. This heat is continuously removed from the reactor by the recirculation of pressurized water called "reactor coolant". The main sources of radioactivity to be handled by waste disposal are generated by several processes which occur in the Reactor Coolant System during operation of the plant. These processes are described as follows:

(1) Impurities Introduced by Corrosion - Experience indicates that the corrosion of the Reactor Coolant System surfaces will be on the order of a few hundred-thousandths of an inch per year. This amount of corrosion within a conventional power plant would be insignificant. However, in a reactor power plant, such as the PWR, the minute quantities of corrosion products which become suspended in the coolant will be activated in passing through the reactor core and constitute a source of induced radioactivity. The corrosion product is primarily solid iron oxide, but it also contains oxides of the other elements of stainless steel. In neutral water a large fraction of the corrosion product appears as very fine particles (called "crud") which are partially transportable in the coolant. In alkaline or high pH water corrosion is reduced, the corrosion product appears mainly as a tenacious film on stainless steel, and far lower crud concentrations are produced in the water. In either case the radioactive crud appears as a contaminant in coolant water which may be discharged to the waste disposal system; it is also a contaminant encountered in maintenance and component decontamination operations.

(2) Tritium Formation - Because corrosion is reduced in high pH water, the reactor coolant is treated with lithium hydroxide to raise the pH of the

water to a range of 9.5 to 10.5. Lithium, however, reacts with neutrons generated by the reactor to yield radioactive tritium (heavy-hydrogen-three). Tritium thus appears as a contaminant in coolant water that may be discharged to waste. The tritium having a 12-year half-life decays very little during any reasonable hold-up time and cannot be removed from the coolant water by practical methods. However, it has a biological half-life of only 19 days and emits a very weak beta radiation and, therefore, relatively high concentrations are permitted in drinking water (see Table III); thus, disposal can be accomplished readily by dilution and dispersion. For instance, it is estimated that one part of reactor coolant with the highest anticipated tritium content can be reduced to one-tenth of the lifetime drinking water tolerance limits by mixing with 25 parts of ordinary natural water.

(3) Fission Products - Another way in which the reactor coolant can become contaminated is by fission products released by the core itself. The PWR core utilizes as fuel both enriched uranium (U^{235}) as a uranium-zirconium alloy and natural, or unenriched, uranium dioxide (UO_2) in zirconium-clad rods. Both types of fuel elements are made with high integrity zirconium cladding to contain the fuel and its fission products. However, despite rigid manufacturing inspections and closely controlled manufacturing techniques, the possibility exists that a small percentage of the UO_2 fuel elements in the PWR core might develop small leaks in the cladding and a portion of the fission products contained therein would be released to the coolant. It was assumed as a conservative basis of design that 1 percent (1000) of the UO_2 elements would develop small defects and that at any time the plant should be able to operate with these 1000 defective elements.

The waste disposal system is therefore designed to handle gaseous and soluble fission products.

(4) Control of Radioactivity in the Water - The three processes described above introduce contaminants into the coolant water. To limit the buildup of these contaminants in the water, the coolant that passes through the nuclear core is continuously purified by circulating a portion of it through a bypass demineralizer resin which removes both insoluble matter and soluble impurities. Because of the nature of the resin, the lithium and tritium are not removed, but the resin does collect and concentrate crud and fission product radioactivity from the coolant. Upon saturation the resin is not regenerated but is transported to buried storage tanks in the waste disposal plant.

The above paragraphs describe the prime sources of radioactive waste in the PWR plant. It should be noted that no facilities for chemically processing spent nuclear fuel are planned for the PWR site. The radioactivity of the PWR wastes is relatively low, and the problems of waste disposal at Shippingport are therefore considerably less than those at a fuel processing site.

DESIGN DISPOSAL CRITERIA (Ref. 4)

The problems associated with the disposal of radioactive wastes at an atomic power plant situated in a populated area are much the same as those encountered in the disposal of chemical or sanitary wastes. As is the case for chemical or sanitary wastes, maximum permissible concentrations (MPC) for radioactive pollutants in air and drinking water have been established (Refs. 5 and 6), and a major aspect of the design of the radioactive waste disposal facilities at an atomic power plant is to insure that these tolerance values are not exceeded.

An example of the maximum permissible concentrations in air and water for some typical radionuclides is given in Table III.

Specific PWR Criteria

The following criteria have been established for the basis of design of the PWR waste disposal system:

1. Before dispersal to the air or the river, any radioactive waste fluids produced by the PWR plant must be reduced in concentration, when necessary, by a plant waste disposal system to a tolerable concentration: (a) in water, measured at the condenser water stream before discharge to the Ohio River and (b) in air, measured at the point of discharge to the surrounding atmosphere. A "tolerable concentration" is defined as one which would not cause an increase in the background radioactivity of more than one-tenth of the maximum permissible concentrations (MPC) for drinking water and air recommended in Table 5, National Bureau of Standards Handbook 61, entitled "Regulations of Radiation Exposure by Legislative Means" (Ref. 6). One-tenth of the listed MPC is used for design in order to be conservative and because if, in the future, some permissible limits were to be lowered, the provision of additional disposal facilities might be difficult and expensive.

Because data on the quantity and activity of all the specific radioisotopes in the PWR coolant were not available during the design stages of the radioactive waste disposal plant, the design for disposal of nonvolatile waste material was based on one-tenth of the value listed for an unidentified mixture of radioisotopes as shown in Table 5 of NBS Handbook 61 (Ref. 6). In general, the MPC for a mixture of unidentified isotopes is more restrictive by a factor of 100 than the corresponding value for specific isotopes.

2. In addition, any released material must conform to the requirements given for disposal of radioactive wastes in "Codes of the State of Pennsylvania" (Ref. 7).

3. Any tank or buried storage facility must be protected against leakage into the ground and vapor release into the atmosphere.

4. Sufficient capacity must be provided to store a maximum expected quantity of radioactive gases for at least the maximum duration of a weather inversion. *

5. Any wastes that are shipped from the site must be packaged in a manner to prevent a radioactive hazard to personnel involved in the packaging or shipping operations. Regulations as set forth in the Code of Federal Regulations (Parts 71 to 78) for public carriers will be observed (Ref. 9).

METHODS CONSIDERED AND SELECTION FOR PWR

Several different methods of disposing of the various types of radioactive waste material are in use at the National Laboratories and other Atomic Energy Commission (AEC) installations. Many other methods have been suggested and proposed in AEC documents. In general, these methods can be classified into the following types:

1. Natural Decay
2. Dilution and Dispersion
3. Land or Sea Burial
4. Concentration and Storage

An extensive study was undertaken to determine which methods were

* The maximum expected duration of a weather inversion in the area is five days, based on the Donora incident (Ref. 8).

best suited for the specific requirements of the PWR plant. The conclusions of this study are summarized below.

Liquid Waste Disposal

Decay

The over-all reduction in activity level required for the nonvolatile (nongaseous) fission products in the PWR wastes before they can be discharged to the environment was postulated as approximately 2×10^8, based on the design criteria of operating the plant with 1000 defective fuel elements. It was apparent that this factor could not be achieved in a practical manner by natural decay processes alone. Based on design values for the long-lived radioisotopes, 33 yr Cs-137 and 20 yr Sr-90, the average hold-up time required to reduce the activity of PWR wastes to one-tenth the MPC was predicted to be approximately 80 years. Table I illustrates the predicted rate of gross decay for the assumed nonvolatile fission products in the PWR wastes.

Direct Dilution

By proper blending with copious amounts of uncontaminated water, radioactive liquid wastes may be disposed of without exceeding permissible tolerance levels. Provided that an adequate supply of uncontaminated water is available, this method is perhaps the least expensive and requires minimum handling.

Although the PWR plant is situated advantageously on a fairly large river, it was decided early in the design stages of the PWR plant

not to rely on the Ohio River proper for the dilution of liquid wastes but to use only the quantity of river water brought into the plant to cool the main condenser. This decision was made because of the impossibility of obtaining complete mixing in the river with the consequent chance of exceeding waste disposal specifications and because of the general philosophy of not using the entire stream for dilution because of the possibility of other nuclear plants being located on the river in the future. By properly controlling the discharge of low-level wastes, the limited dilution process utilized is entirely under the close supervision of the station operator who is provided with suitable alarms and instruments to protect against accidental discharge. Furthermore, the river, which provides some additional dilution, serves as an additional design safety factor.

Disposal by Ocean Burial (Ref. 1)

Disposal of all radioactive liquid wastes by ocean burial was explored. The procedure, similar to that used by other AEC installations and modified to suit the special conditions of PWR, would entail introducing a quantity of waste liquor as received from the reactor plant together with dry cement into a 55-gal steel drum, mixing the contents by rolling the drum, and then allowing the mixture to solidify. The drums would then be trucked to a marine terminal and picked up by a vessel which would complete the ocean burial.

Because of large quantities of liquids handled at the PWR, ocean burial without concentration was found to be too expensive -- $3.50 to $10.00 per gallon with as much as 30,000 gallons to be drummed each month.

Evaporation

A successful means of disposing of radioactive waste liquors in large-scale operations is by evaporation concentration, with land or ocean burial of the radioactive residue liquor. This is the practice at various AEC installations.

Practical, proven designs have been developed for evaporators in which the activity of the distillate is less than one-millionth of the activity of the bottoms liquor (i.e., decontamination factor = 10^6). Essentially this has been achieved by the design of highly efficient deep bed vapor filters for the removal of entrained liquid particles.

In spite of the merits of evaporation, it is relatively costly ($0.10 to 0.13 per gallon) and complicated. An example of the complexity of the process is the need for remote disassembly of the evaporators and replacement of the reboiler sections. Maintenance of equipment or piping in contact with high activity liquid would be complicated, even for minor repair work.

For these reasons the use of evaporation for disposal of all liquid wastes was ruled out. However, a limited vapor compression evaporation process seemed most feasible for processing wastes containing large amounts of dissolved solids. As described in a later section, this liquid cannot be processed through the ion exchange system without sacrifice of resin or resin bed performance.

Montmorillonite Clay (Ref. 2)

Certain clays exhibit a cationic exchange capacity. The exchange efficiency depends upon the type of clay used and the chemical composition of the waste solution. The Brookhaven National Laboratories have pioneered the use of montmorillonite clay for this purpose based on its superior ion exchange and mechanical stability when extruded into columns. The adsorption of radioactive ions from an aqueous solution onto montmorillonite clay, followed by a high-temperature sintering operation (1000°C) which locks the adsorbed material into a nonleachable state, is attractive. The sintered product could be disposed of in the ocean or land without danger of dissemination of the contained radioactivity.

Although this process had been successfully demonstrated, at the time of the initial design of the radioactive waste disposal facilities of the Shippingport plant it was concluded that the use of this process would require a great deal of developmental and pilot plant work to develop it into a routine commercial procedure.

Ion Exchange (Ref. 3)

Decontaminating liquid waste simply by passing it through ion-exchange resins has been the basis for much AEC-sponsored research. Resin exchange has proved useful in purification of pressurized water and is being incorporated for similar service in the PWR reactor system.

The effectiveness of ion-exchange resins in removing fission product radioisotopes from various laundry and chemical process wastes has been established. However, experimental data were lacking for the decontamination factors that could be achieved and the amount of waste that could be processed before the resin would saturate. Because of the merits of ion exchange, a

research program was instituted at Bettis Plant to obtain this information. Results of a research program, consisting of in-pile tests of defective PWR-type fuel elements, demonstrated that the decontamination factors obtained on typical PWR wastes are at least as great as required by the PWR Waste Disposal System design. Greater factors than required are very probable.

In addition, it was found that large volumes of reactor quality water, containing the design levels of radioactivity, could be processed by ion exchange with reasonable resin consumption.

As a result of these and earlier studies made by Bettis Plant, ion exchange was chosen as a principal method of high-purity liquid waste disposal at the PWR supplemented in particular instances by other methods as discussed below.

Solid Waste Disposal

Spent Demineralizer Resin

One of the methods considered for disposal of the spent demineralizer resin was burning the resin in an incinerator. Apparent disadvantages include (1) the incinerator would require heavy shielding to protect operating personnel, (2) probably the most important, the radioactive gases from the incinerator would require special washing and treatment before discharge to the atmosphere, and (3) intermediate storage of high-activity resin would be required. For the reasons given this method was not considered applicable to PWR.

Flushing spent resin from the demineralizer directly into an underground storage tank, without the use of either remote handling or other equipment, was also considered. The resin would be allowed to settle and the supernatant liquid above the resin-water interface removed for further processing.

This storage tank method of resin waste disposal was selected for PWR because it is the simplest, safest, and most economical in capital and operating costs.

Combustible Waste

Disposal of combustible waste by either immobilization in concrete, ocean or land disposal, or incineration was studied. Ocean disposal requires baling dry waste with a standard baling machine, placing the bale in a 55-gal drum, casting the bale in concrete, and transporting the drum to a suitable point in the ocean for disposal. Although simple baling gives a volume reduction of only 6 to 1, it has the advantage of no appreciable contamination of air about the baler. Based on a 12 in. by 12 in. by 30 in. bale, the largest bale from a standard baling machine that will fit into a 55-gal drum and allow sufficient concrete shielding, it was estimated that the cost was $0.45/lb of waste.

Land disposal was considered undesirable because of the nonavailability of suitable burial grounds. To establish a new burial ground would require extensive geological investigations and the establishment of perpetual monitoring of the area.

The third method, incineration, would involve capital cost of equipment much larger than for baling and ultimate disposal in the ocean but would give a much larger over-all reduction in volume (50 to 1). For this method the total cost per pound of waste was estimated to be $0.30.

Incineration was selected as the method to be used at PWR for combustible waste disposal, because it is more economical than baling and ocean burial, the only other disposal method considered practical.

Gaseous Waste Disposal

Dilution and Dispersion

The generally accepted method of disposing of radioactive waste gases is by dilution with air and discharge to the atmosphere from a high stack. This is standard procedure at AEC installations where fission-product gases are released during processing of spent fuel elements. Since the PWR is located in a populated area, disposal of waste fission-product gases by air dilution alone and discharge to the atmosphere would require, under the conservative design assumptions, a volume of air so large that it would be impractical.

Liquefaction

Liquefaction is a disposal method which was considered, but it was still in the pilot plant stage at the time of the design of the PWR system. This method consists of liquefying the fission gases and would require expensive low-temperature equipment.

Decay and Dilution

Because the fission gases from the PWR plant are principally 5.3-day Xe-133 and, to a lesser extent, 10-yr Kr-85, a practical solution for the disposal of these waste gases was found to be by decay and dilution. By allowing the waste gases to decay for a period of 60 days, a 2500-fold reduction in their activity could be realized. This reduction, coupled with a reasonable quantity of air dilution, would allow the gases to be reduced to the allowable tolerance. Table II illustrates the predicted rate of decay for the volatile fission products in the PWR wastes.

COMMONWEALTH OF PENNSYLVANIA INDUSTRIAL WASTE PERMIT

Industrial Waste Permit, #1832, issued by the Pennsylvania Department of Health Sanitary Water Board to the Duquesne Light Company, limits the liquid river discharge (exclusive of tritium) to 1,590 microcuries per day (μc/day) averaged over any 365-day period, with a maximum of 6,200 microcuries for any one day. The 1,590μc/day was conservatively selected on the basis of the originally expected radioactivity levels at the PWR. The 6,200μc/day is the maximum quantity of liquid radioactive wastes that can be diluted by the 114,000 gpm flow of both condenser cooling water pumps and still meet the criterion that the resultant increase in concentration does not exceed one-tenth of the MPC (1×10^{-8} μc/cc). The corresponding tritium release is limited to 10 curies per day average with a maximum of 50 curies per day. Both of these limits were conservatively selected on the basis of the maximum activity expected in the effluent. The 10 curies/day corresponds to a concentration of only 0.01% of the MPC in the effluent stream. Since all radioisotopes are not equally hazardous, the wide range in allowable discharge between an unidentified mixture of beta-gamma emitters and tritium is to be expected.

DESCRIPTION OF PROCESS

The radioactive wastes that are to be treated at the PWR plant can be generally classified as solid, liquid, and gaseous wastes. The liquid wastes can be further classified as (1) reactor plant effluents, (2) monitored wastes, (3) special monitored wastes, and (4) decontamination room wastes. The solid wastes can also be subdivided into combustible solid wastes and noncombustible solid wastes. Each classification of waste requires a somewhat different waste treatment process. These processes are described

in the succeeding paragraphs. As previously mentioned, several basic waste treatment processes are used in the HWR plant, depending on the type of waste. These processes are (1) natural decay of the radioactive constituents, (2) ion exchange to remove the radioactive constituents, (3) dilution in the condenser effluent stream, (4) separation of radioactive material from waste water by a standard evaporation process, (5) retaining certain solid wastes at the plant site in underground storage tanks, (6) incineration of certain solid wastes, and (7) packaging of wastes for land or sea burial. Some wastes use only one of the above processes, while others require combinations of these processes.

Reactor Plant Effluents

The waste effluent (reactor coolant) from the reactor plant may contain radioactive soluble and insoluble materials as well as dissolved radioactive gases. The estimated average total volume of these wastes is 23,000 gals/mo. (See Fig. 1).

The treatment of these reactor plant effluents is (1) decay, followed by (2) ion exchange, and finally (3) separation (and decay) of the dissolved gases from the liquid. The gaseous or volatile activity of the waste is estimated to be a maximum of 10.6 microcuries per milliliter (μc/ml) of liquid, whereas the nonvolatile activity is estimated to be a maximum of 2.9 μc/ml. After treatment the radioactivity of the waste is estimated to be a maximum of 4.5×10^{-5} μc/ml for the nonvolatile and 1.2×10^{-8} μc/ml for the volatile activity. The liquid waste is monitored in the system's test tanks and the proper flowrate determined so that, after blending with the turbine condenser cooling water stream, the activity on leaving the plant will be no more than 1×10^{-8} μc/ml (which is one-tenth of the maximum permissible concentration for an unidentified mixture of isotopes as shown in Table 5 of NBS Handbook 61).

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In order to assure that the desired degree of treatment is obtained, the liquid wastes are sampled and the radioactivity determined at each stage of the process. The entire treatment is a series of batch processes, and the waste may be reprocessed if the samples indicate that the desired radioactivity reduction is not accomplished.

The reactor plant effluents are first transported to hold-up tanks known as "surge and decay tanks". The surge and decay tanks are designed to provide storage volume for the reactor plant effluents for 45 days if necessary. For this maximum storage time, design estimates indicate that the radioactivity content will be decreased by a factor of 31 for the nonvolatiles and by a factor of 360 for the volatiles. Actual storage time is determined by the extent of radioactive decay required and in most cases will be less than 45 days. Each tank is approximately 23 ft in diameter by 12 ft high, with an operating capacity of 29,100 gals. The tanks are constructed of type 347 stainless steel and are completely below ground level. The tanks have been located below grade in order to provide the necessary shielding against radiation.

The waste liquid is sampled in the surge and decay tanks to ensure that adequate reduction of radioactivity has been accomplished before further processing is initiated. If it is necessary to further reduce the radioactivity, the next treatment for this liquid is in ion exchangers. Laboratory data has shown that the nonvolatile radioactivity will be reduced by at least a factor of 1000 in ion exchangers of the type used in the waste disposal system. The volatile activity is not appreciably reduced in these demineralizers.

The liquid from the surge and decay tanks is first passed through a heat exchanger whose function is to cool this liquid so that the capacity of

the ion-exchange resin will not be depleted by hot liquids. The liquid then flows through a series of four ion exchangers so manifolded that the relative position of any one ion exchanger in the series may be changed to gain optimum operating characteristics. The liquid leaving the ion exchangers will have had its nonvolatile activity reduced sufficiently, but the liquid may still contain a high level of volatile radioactivity.

The ion exchangers are located below ground level in the waste disposal area. Shielding required for personnel protection is provided by the below-grade location and also by concrete enclosures for the ion exchangers. The ion exchangers are about 2 ft ID by 6 ft 6 in. high and contain a 4 ft deep resin bed. The ion exchanger vessels are fabricated of type 347 stainless steel.

The waste liquid is then sent to a gas stripper where almost all of the radioactive gases are removed from the liquid. The design and operation of this stripper will be described in the gaseous waste section. The activity due to volatile content of the liquid is reduced by a factor of 2×10^6 in the stripper, while the nonvolatile activity is not significantly affected.

Liquid waste from the stripper is then put in one of two available test tanks. Each test tank is located at ground level, has an operating capacity of 5000 gals, and is fabricated of carbon steel and lined with a suitable plastic material. Shielding is not required, since the radioactivity has been reduced to a very low level. The function of these tanks is to contain a processed liquid until analyses can be made to determine whether the desired reduction in activity has been obtained. Normal operations require that one of the test tanks is being filled while the other test tank is, after satisfactory analyses have been obtained, being emptied to the

turbine condenser discharge stream. The waste liquid is pumped to this stream through a flow regulator and activity monitor, and the flowrate and activity are continuously recorded.

The waste liquid is discharged into the condenser cooling water through a distributor located in the plant effluent channel. The blended liquids then pass over a weir where further blending is accomplished. In the effluent channel downstream of this weir is located a sampling rake which obtains samples from 21 different locations in the channel. These samples are mixed and a representative sample collected to determine that the activity of the liquid leaving the plant meets the criteria established in the plant's radioactive waste disposal permit.

Monitored Wastes

Monitored wastes are the effluents from the showers and cold (non-radioactive) laundry. This effluent is not expected to be radioactive under normal conditions but is handled in the waste disposal system as a safety precaution.

The monitored wastes are pumped from the service building into tanks located in the waste disposal area. The estimated volume of this waste stream is 170,000 gals/mo, and the estimated (maximum) radioactivity (all activity is due to nonvolatiles) is 7.0×10^{-6} μc/ml. The liquid from these tanks is monitored for radioactivity in the laboratory, and the liquid is discharged to the effluent channel where it is diluted with the turbine condenser effluent stream prior to discharge to the Ohio River.

Special Monitored Wastes

The special monitored wastes are the effluents from the "hot" laundry and laboratory drains. The activity of this stream may be considerably greater than that of the monitored wastes.

The special monitored wastes are pumped from the service building into tanks that are also located in the waste disposal area. The estimated volume of this waste stream is 89,000 gals/mo, and the estimated maximum activity (all nonvolatile activity) is 1.70×10^{-4} μc/ml. The liquid from these tanks is monitored for radioactivity in the laboratory and then pumped to the effluent channel at a controlled flowrate such as to produce a blended activity of 10^{-8} μc/ml or less. If the activity should be so great that the tank cannot be emptied in the normal operating time, the liquid may be transferred to the decontamination room waste tanks for evaporation.

Decontamination Room Wastes

The decontamination room wastes are the effluents from the decontamination or cleaning rooms.

These wastes are also pumped from the service building into the tanks in the waste disposal building. The estimated volume of this waste stream is 7,700 gals/mo, and the estimated maximum activity is 0.27 μc/ml. This stream may contain a high level of dissolved solids, and the method of treatment will be to concentrate these solids by evaporation, mix the concentrate with cement in standard 55-gal drums, and then to dispose of these drums by burial in the ocean. The decontamination room waste stream is first neutralized with caustic soda or sodium carbonate and then fed into a vapor compression type evaporator. The evaporated liquid (overhead) is sent to the surge and decay tanks where it is processed as part of the reactor plant effluents. This is merely a safety precaution to ensure that high activity level liquid is not discharged to the effluent channel. Actually, the overhead product is expected to be essentially free of radioactivity as has been demonstrated in other AEC installations. The concentrated liquid in the evaporator is allowed to accumulate until sufficient

material has been collected to fill a drum. The material is then pumped into a drum containing dry cement. The drum is then rolled to insure a good mixture and the cement allowed to harden. The drum is eventually shipped to the coast for ocean burial.

The evaporator unit is of stainless steel construction and has a capacity of 100 gals/hr. (See Fig. 4.) This unit is of the standard vapor compression type and is located in a shielded vault alongside the waste disposal building. The vault in which the evaporator is located is not normally occupied, since all evaporator operations are performed remotely.

Gaseous Wastes

Minute quantities of radioactive gas, primarily xenon^{133} and krypton^{85}, along with hydrogen, are brought into the radioactive waste disposal plant with the reactor plant effluent. A portion of these gases is released as hot reactor coolant is discharged to a flash tank, which is designed to quench the flashing liquid as it is depressurized. The vapor spaces of this and other tanks where reactor coolant is stored are continuously purged with a recirculating flush gas stream. This closed gas system serves to contain the radioactive gases and control the buildup of hydrogen. The hydrogen content in the flush gas stream is maintained below 1% by passing it through a catalytic hydrogen burner where the hydrogen is combined with oxygen to form water. This prevents the formation of an inflammable hydrogen-air mixture (4% hydrogen) in the waste disposal system.

Some gases remain in the liquid and are transported from the flash tank to the large underground surge tanks provided for the storage of the liquid

and decay of the radioisotopes. The waste water in the surge tanks is sampled periodically to determine when adequate reduction of the radioactivity has been accomplished. The waste liquid is next processed through the ion exchangers to further reduce the nonvolatile activity. After passing through the ion exchangers, the waste liquid is sent to the gas stripper where almost all of the remaining gases are removed. Steam passes up through the stripper which operates under a slight vacuum and reduces the gas concentration in the falling liquid to an insignificant amount. The stripped gases are sent to the circulating flush gas system.

At infrequent intervals gas is discharged from the flush gas system into one of four decay drums where it is stored at 50 psig. This gas is stored for a sufficient period of time (60 days design) for the radioactivity level to decrease to a point where subsequent controlled discharge and dilution with 9000 cfm of air from standard industrial fans will permit discharge of the gases at less than one-tenth of the maximum permissible concentration as specified in the NBS Handbook 52.

The gas system provides a "breathing system" for all tanks in the waste disposal plant. This completely enclosed system has enough surge capacity to receive and contain all gaseous discharges from the reactor plant and to maintain a constant pressure in the vapor spaces of the tanks in the waste disposal plant as they "breathe" due to changes in liquid level.

Combustible Solid Wastes (See Fig. 2)

Combustible solid wastes are transported to the waste disposal building where they are burned in an incinerator. The flue gases are sent through a

wet gas scrubber in which particulate matter 5 microns and larger is removed from the gas. In addition, flue gas is also cooled in the wet gas scrubber. The cooled flue gas is then passed through an exhaust filter for final clean-up before discharge to the stack. In the original design the residue ash was mixed with water and pumped to an underground storage tank. Operational difficulties led to the development of a procedure for direct drumming of the residue ash for ocean burial. It is estimated that 2000 to 4000 lbs of combustible waste may be processed each month. No realistic estimate can be made of the activity of this waste, since it may vary over an extremely wide range.

Noncombustible Solid Wastes

The noncombustible solid wastes consist of resins from the plant's ion exchangers, residue ash from the incinerator, solids from strainers in the pipe lines, and contaminated items of plant equipment. The method of disposing of this last category is to imbed the equipment in concrete and bury it at sea if the item is relatively small. If a large piece of equipment is involved, it will be imbedded in concrete and stored at the site.

Contaminated resins and strainer solids are disposed of by slurrying with water and then pumping the slurry into one of two resin storage tanks. The solids are allowed to settle in these tanks and the supernatant water is educted into one of the surge and decay tanks. This liquid is then processed in the same manner and by the same equipment as the reactor plant effluents.

The estimated volume of solid, noncombustible material that will be placed in the resin storage tanks is a maximum of 36 ft^3/mo. The water required to slurry this resin has been estimated at 2800 gals per mo. This residual

water contains some radioactive material.

The resin storage tanks are located in the waste disposal area in individual, totally enclosed, underground vaults several feet below the ground level. Each tank is approximately 10 ft high by 15 ft in diameter with a net operating capacity of 11,600 gals. The tanks are constructed principally of type 347 stainless steel. Shielding of these tanks is provided by the concrete enclosure and the earth cover.

The incinerator and ash drumming equipment are located in the waste disposal building. This incinerator is fabricated of stainless steel and requires no special shielding, because it is remotely operated and is located in a pit in the building. The incinerator is approximately 8 ft 5 in. high by 3 ft 4½ in. OD. The drumming equipment consists of a hydraulic jack used to support a 55-gal steel drum under the incinerator dump valve and the necessary drum handling equipment. Concrete is added to the ash-filled drums prior to shipment for ocean burial.

ULTIMATE DISPOSAL OF RADIOACTIVITY

Approximately 80% of the radioactivity delivered to the waste disposal system from the reactor plant is due to the presence of fission and corrosion products in the reactor coolant. Under design conditions 99.34% of this activity will be dissipated by natural decay or collected on ion exchange resin before the wastes are discharged to the environment. Only 0.17% will be discharged to the atmosphere, 0.0034% to the Ohio River, and the remaining 0.4866% will be contained in the wastes which are drummed for ocean burial.

The remaining 20% of the radioactivity delivered to the waste disposal system from the reactor plant is due to the presence of tritium in the reactor coolant. Dilution and dispersion of tritium in the waste effluent stream adequately reduces the tritium concentration to less than 1/100 of the tolerance level.

WASTE DISPOSAL PLANT PERFORMANCE

Liquid Effluents (See Table IV)

The predicted average monthly accumulation of reactor plant liquid effluents was 23,000 gallons. In the first eight months of operation, from 2 to 14 times the design quantity was discharged per month. Special operating conditions, necessitated by the extensive testing program carried out during this period, accounted for most of this excess. The storage capacity provided in the Radioactive Waste Disposal System was determined from a correlation of both quantity and level of activity of the liquid to be handled. The activity level of the liquid received during this period was low enough to permit immediate discharge to the river at a controlled rate. For this reason the excess volume has not been a problem. It is of interest to note that during the months of June, July, August, and September of 1958 the quantity of reactor plant effluents became steady at about three times the design volume, principally because the mode of reactor plant operation was more nearly typical of a central station. During this period of continuous operation the source of the major portion of excess liquid was determined. Approximately 12% of the excess was the result of the increased sampling of primary coolant necessitated by the test program. Another 12% was the result of frequent loop drainage for inspection of the heat exchangers and boilers as a part of the initial testing program. Plant leakage accounted for about 55% of the total excess. An extensive program was initiated to remedy this condition. Particular emphasis was placed on reducing leakage through reactor coolant system relief valves, which were known contributors. However, the waste disposal plant was designed to accommodate this excess volume.

Ion Exchangers and Gas Stripper

Thus far, there has not been a continuing need for the use of the ion exchange and gas stripping equipment provided for processing radioactive liquid. Therefore, verification of the predicted decontamination factors is not yet possible. However, waste liquids collected in the surge tanks were processed through a laboratory ion exchanger under loading conditions similar to those in the plant waste disposal system. On these low activity level wastes decontamination factors for nonvolatiles of from 300 to 400 have been obtained. Also, the evaluation of development work on the removal of fission products from high-purity water by ion exchange was recently completed. Average decontamination factors, across a single resin bed, of 1000 were reported for solutions of radioactive fission products of the type and amount possible in the PWR reactor plant effluents. In addition, this average DF was observed over periods of time which corresponded to the processing of at least 3000 column volumes of the solution. This indicates that the original estimate of the resin exhaustion rate (8 cu ft/mo) is about 7.5 times greater than can now be predicted. Considering that four ion exchangers will be used in series for waste liquid processing at PWR and that higher decontamination factors will be obtained at higher activity levels, the design DF of 1000 should prove to be a conservative figure.

The average monthly volume of monitored waste liquids encountered has been approximately 30 percent of the predicted volume of 170,000 gals. Special monitored waste liquid received has been approximately 40 percent of the predicted quantity of 89,000 gals/mo. The activities encountered are below design values. Both volume and activity levels are expected to increase as the degree of plant contamination increases.

Waste Disposal Vent Gas System

Sufficient data are not available to evaluate the design of the Waste Disposal Vent Gas System. However, it is probable that larger quantities of hydrogen than the predicted 70 scf/mo have been handled thus far because of the abnormal operating conditions of the testing program and higher than expected relief valve leakage. About 3740 standard cubic feet of low-activity level gases were discharged to the atmosphere from the system during the eleven-month period. The total activity discharged during this period was less than 10% of the permissible discharge for a single month. Because of the low activity level, a special bypass was arranged to allow discharge at rates up to 3 scfm. The permanently installed flow controller has a maximum capacity of 1.5 scfh. The vent gas system has functioned properly in maintaining a constant system pressure, and the hydrogen content of the system has been maintained well below the explosive range at all times.

Incinerator

The amount of combustible solid wastes disposed of in the incinerator has approximated the predicted average of 4000 pounds per month. In general, all disposable combustible items used in an area of possible contamination have been burned. The average activity level per incinerator charge has been near the lower limit of measurability.

Total Liquid Waste Discharged

The criteria for the discharge of radioactive wastes are contained in a permit granted to the Duquesne Light Company by the Commonwealth of Pennsylvania. Under the terms of the permit a maximum of 6200 μc/day and a

yearly average of 1590 μc/day of radioactivity may be discharged to the Ohio River. Because of its high tolerance level, tritium activity is determined separately and has an individual discharge limit of 300 curies per month. The amount of radioactivity discharged in waste liquid during the first few months of operation is shown in Table IV. It should be noted that the total amount of radioactivity discharged to date is less than that permissible during a single month. This is an indication of the margin available for safe disposal of liquids at the higher activity levels expected near the end of the reactor core life.

CONTROL OF THE PROCESS

System

As mentioned above in the description of the waste treatment processes, provisions are made to sample the waste streams as they enter the waste disposal system, before and after each waste treatment process, and before discharge from the plant. Sampling facilities are also provided so that diluted waste streams, both liquid and gaseous, can be monitored after dilution. The system provides flexibility in that any waste stream may be reprocessed if necessary to reduce the radioactivity to a level where subsequent dilution will yield a stream at or below the maximum permissible concentration chosen for discharge from the plant. This flexibility, along with the basic batch type of processing in the system and the large number of sample locations, will minimize the possibility of discharging wastes from the plant which are "over-tolerance".

In addition to the numerous sampling provisions provided for the system, effluent activity monitors are installed in the main liquid effluent header and on the stack. These monitors will detect and inform the operator of accidental discharge of either liquid or gaseous wastes that are over-tolerance.

Off-Site Monitoring Program

An off-site monitoring program was started early in 1956 and has continued during operation of the plant. Its purpose has been to determine the types and amounts of radioactive materials that occur naturally in the environment around the PWR plant site; the program was started approximately 1½ years prior to startup of the nuclear portion of the plant. The pre-operational data have permitted a good evaluation of the environmental radioactivity so that studies now being conducted after the plant is in operation will reveal any significant changes that might be due to plant operations. Evaluations have been and are being made on: (1) soil in the general vicinity of the plant; (2) the Ohio River water, above and below the site; (3) well water within a one-mile radius of the site; (4) vegetation in this general area; and (5) the air in this general area.

River water analyses are being obtained from two locations above and one below the site. Another monitoring station is currently being installed at the "first point of use" below the plant, which is the water-intake service to the city of Midland, Pa.

To properly evaluate the presence of radioactive dusts and gases in the air and to measure the normal background radiation in the vicinity, five monitoring stations are being operated. Four of these are fixed stations, while the fifth is mobile. Prevailing wind direction and velocity determined the four fixed locations. These are placed to indicate as accurately as possible the concentrations of radioactivity in the air both downwind and upwind of the plant site. The mobile station will move from location to location to determine the relationship between the four fixed stations and other areas that may be of interest.

In general, averages of gross beta activity in Ohio River water samples both before and after reactor operation have ranged from less than five micro-microcuries per liter to slightly greater than thirty micro-microcuries per liter for composite samples analyzed on a weekly basis. Of these radioactivity levels from 25% to 100% of the gross beta activity has been due to potassium-40, a naturally occurring radioisotope. The results from the two downstream sampling stations have not varied significantly from the upstream station results. The downstream results have fluctuated, giving results that were in some cases lower and in some cases higher than the upstream data. These fluctuations have shown no correlation with radioactive liquid wastes discharged to the river.

Only minor quantities of radioactive gases have been released from the Radioactive Waste Disposal System since plant operation began. These gases were principally of the noble gas family, i.e., xenon or krypton, and no evidence of these discharges has been detected by the area monitors.

Since PWR went critical in December of 1957, the results of the environmental radiation survey have shown no indication of an increase in area radioactivity levels that could be attributed to operation of the Shippingport plant. Such a result is to be expected, based on the known levels of radioactivity actually discharged to the environment from the reactor plant and discussed earlier in this paper.

This work was initiated by the AEC in cooperation with local, state, and national public health agencies. It is now being continued as a cooperative effort by the Pennsylvania State Department of Health and the Duquesne Light Company.

The results of all these tests make it possible to determine the adequacy of the waste disposal facilities. Continued operation of the plant is expected to prove the degree of conservatism which has been built into the plant and will provide data that will establish modified criteria for the design of future radioactive waste disposal facilities for other nuclear power plants.

TABLE I

PREDICTED GROSS DECAY FACTORS
FOR NONVOLATILE PWR WASTES

Decay Intervals	Gross Decay Factor
Shutdown to 1 hr	4
1 hr to 24 hrs	5
1 day to 3 days	1.2
1 day to 11 days	2.2
1 day to 16 days	3.2
1 day to 31 days	10
1 day to 46 days	31
1 day to 61 days	73
1 day to 76 days	99

TABLE II

PREDICTED GROSS DECAY FACTORS
FOR VOLATILE PWR WASTES

Decay Interval	Gross Decay Factor
Shutdown to 24 hrs	Not Known
1 day to 3 days	1.3
1 day to 11 days	3.7
1 day to 16 days	7.1
1 day to 31 days	50
1 day to 45 days	360
1 day to 60 days	2500
1 day to 76 days	16,700

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TABLE III

MAXIMUM PERMISSIBLE CONCENTRATIONS OF RADIOISOTOPES (5) (6) (7)

Radioisotope	Microcuries/ml	
	Water	Air
33 yr Cs^{137}	1.5×10^{-3}	2×10^{-7}
19.9 yr Sr^{90}	6×10^{-7}	2×10^{-10}
54 days Sr^{89}	7×10^{-5}	2×10^{-8}
67 hr Mo^{99}	1h	2×10^{-3}
8 day I^{131}	3×10^{-5}	5×10^{-9}
5.3 day Xe^{133}	4×10^{-3}	4×10^{-6}
12 yr H^3	0.2	2×10^{-5}

TABLE IV

LIQUID WASTES DISCHARGED FROM THE SHIPPINGPORT WASTE DISPOSAL PLANT

	Volume (gals/mo)	Nonactive Waste Avg. Activity (μc/ml)	Max. Activity (μc/ml)	Total Activity (μc/mo)
DESIGN	170,000	1×10^{-6}	7×10^{-6}	645
January	33,750	3.4×10^{-7}	1.13×10^{-6}	43.58
February	19,373	6.6×10^{-7}	1.45×10^{-6}	47.78
March	31,928	2.55×10^{-7}	6.39×10^{-7}	30.85
April	11,330	2.95×10^{-7}	4.75×10^{-7}	12.66
May	84,125	2.2×10^{-7}	8.2×10^{-7}	70.11
June	69,440	1.82×10^{-6}	1.4×10^{-5}	478.25

	Volume (gals/mo)	Avg. Activity (μc/ml)	Max. Activity (μc/ml)	Total Activity (μc/mo)
July	55,290	2.2×10^{-6}	6×10^{-6}	473.57
August	72,030	7.1×10^{-7}	1.7×10^{-6}	193.69
September	58,075	6.3×10^{-7}	1.6×10^{-6}	138.08
October	37,500	8.9×10^{-7}	1.9×10^{-6}	127.13
November	71,555	6.43×10^{-7}	1.61×10^{-6}	174.03

Special Waste

	Volume (gals/mo)	Avg. Activity (μc/ml)	Max. Activity (μc/ml)	Total Activity (μc/mo)
DESIGN	89,000	1×10^{-4}	1.7×10^{-4}	32,200
January	35,700	1.7×10^{-7}	2.9×10^{-6}	24.30
February	23,491	1.12×10^{-6}	4.71×10^{-6}	100.40
March	26,279	2.75×10^{-7}	9.65×10^{-7}	27.50
April	20,268	4.8×10^{-6}	2.4×10^{-5}	372.83
May	42,240	3.52×10^{-6}	1.05×10^{-5}	562.15
June	39,670	7.32×10^{-6}	1.7×10^{-5}	1,125.33
July	30,747	8.55×10^{-6}	2.42×10^{-5}	1,019.80
August	46,960	9.76×10^{-6}	3.89×10^{-5}	1,734.31
September	40,940	9.47×10^{-6}	2.17×10^{-5}	1,470.06
October	33,865	6.77×10^{-6}	1.8×10^{-5}	869.73
November	39,410	3.47×10^{-6}	6.63×10^{-6}	518.00

Reactor Plant Effluent

	Volume (gals/mo)	Average Activity (μc/ml)	Maximum Activity (μc/ml)	Total Activity (μc/mo)	Tritium Activity Maximum Curies/day	Tritium Activity Total Curies/mo
DESIGN	23,000	4.5×10^{-5}	1.7×10^{-4}	3,840	50	300
January	87,300	1.89×10^{-6}	5.1×10^{-6}	628.40	3.68×10^{-3}	7.36×10^{-3}
February	45,900	4.69×10^{-7}	1.12×10^{-6}	86.90	0.117	0.813
March	311,027	4.66×10^{-6}	1.46×10^{-5}	5315.50	1.31	6.71

	Volume (gals/mo)	Average Activity ($\mu c/ml$)	Maximum Activity ($\mu c/ml$)	Total Activity ($\mu c/mo$)	Tritium Activity Maximum Curies/day	Tritium Activity Total Curies/mo
April	75,500	9.43×10^{-6}	1.3×10^{-5}	2690.10	5.06	10.06
May	144,340	7.8×10^{-6}	1.8×10^{-5}	3943.10	0.81	3.82
June	60,230	6.8×10^{-6}	9.8×10^{-6}	1588.30	1.56	2.84
July	66,650	1.05×10^{-5}	2.7×10^{-5}	2714.92	0.814	2.26
August	61,510	2.3×10^{-5}	5.83×10^{-5}	5355.45	1.16	4.81
September	67,280	6.4×10^{-6}	1.63×10^{-5}	1482.05	2.10	9.40
October	35,400	1.54×10^{-5}	1.58×10^{-5}	2065.98	1.95	4.98
November	5,000	2.65×10^{-6}	2.65×10^{-6}	50.15	1.73	1.73

Activity Summary

	Nonactive Waste ($\mu c/mo$)	Special Waste ($\mu c/mo$)	Reactor Plant Effluents ($\mu c/mo$)	Total Discharge ($\mu c/mo$)
Allowable (Waste Disposal Permit)				47,700
January	43.50	24.30	628.40	696.20
February	47.78	100.40	86.90	235.08
March	30.85	27.50	5335.50	5393.85
April	12.66	372.83	2690.10	3075.59
May	70.11	562.15	3943.10	4575.36
June	478.25	1125.33	1588.30	3191.88
July	473.56	1019.80	2714.92	4208.28
August	193.69	1734.31	5355.45	7283.45
September	138.08	1470.06	1482.05	3090.19
October	127.13	869.73	2065.98	3062.84
November	174.03	518.00	50.15	742.18

Total Activity Discharged to Date - 35,554.90

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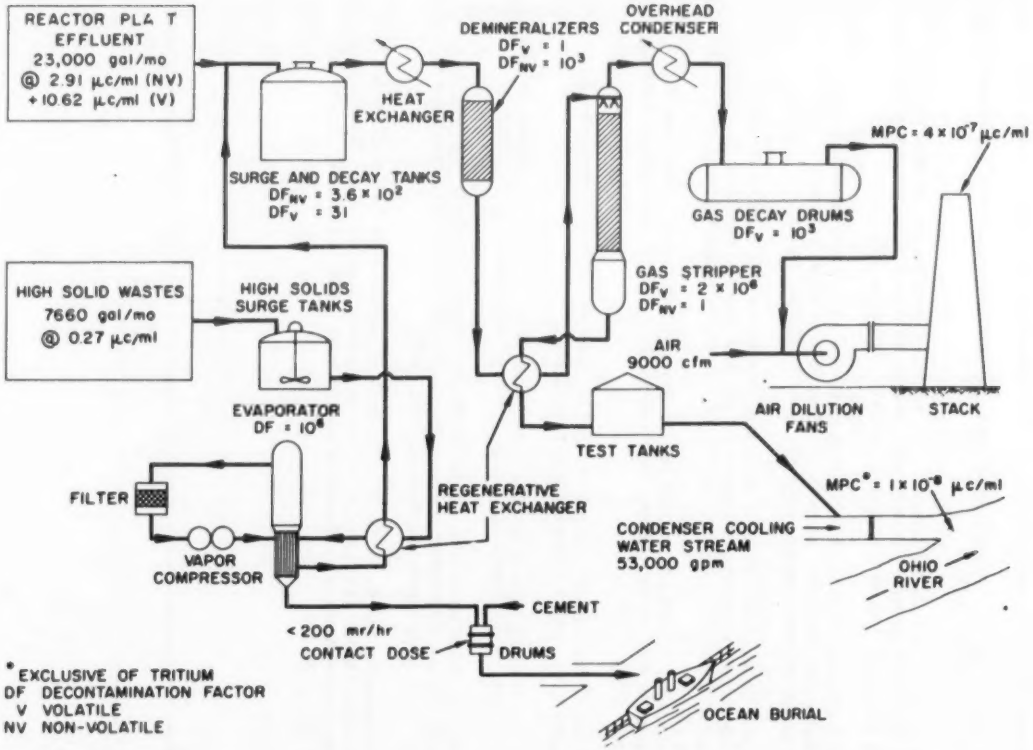


FIGURE 1
RADIOACTIVE WASTE DISPOSAL FACILITIES, SHIPPINGPORT ATOMIC POWER STATION

FIGURE 1
RADIOACTIVE WASTE DISPOSAL FACILITIES, SHIPPINGPORT ATOMIC POWER STATION

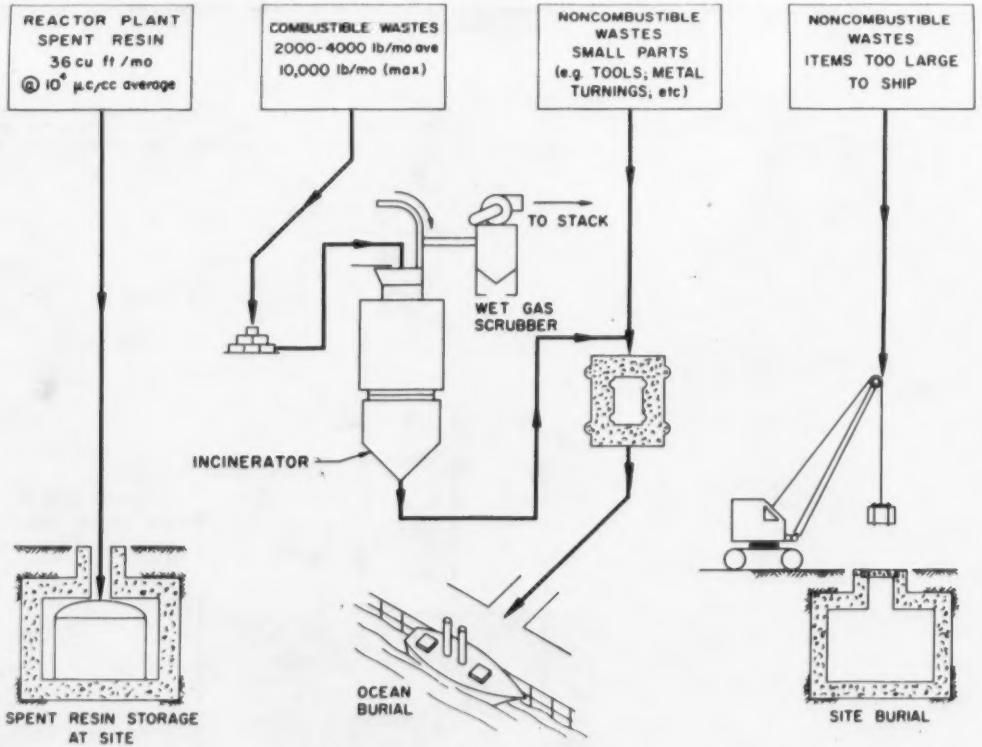
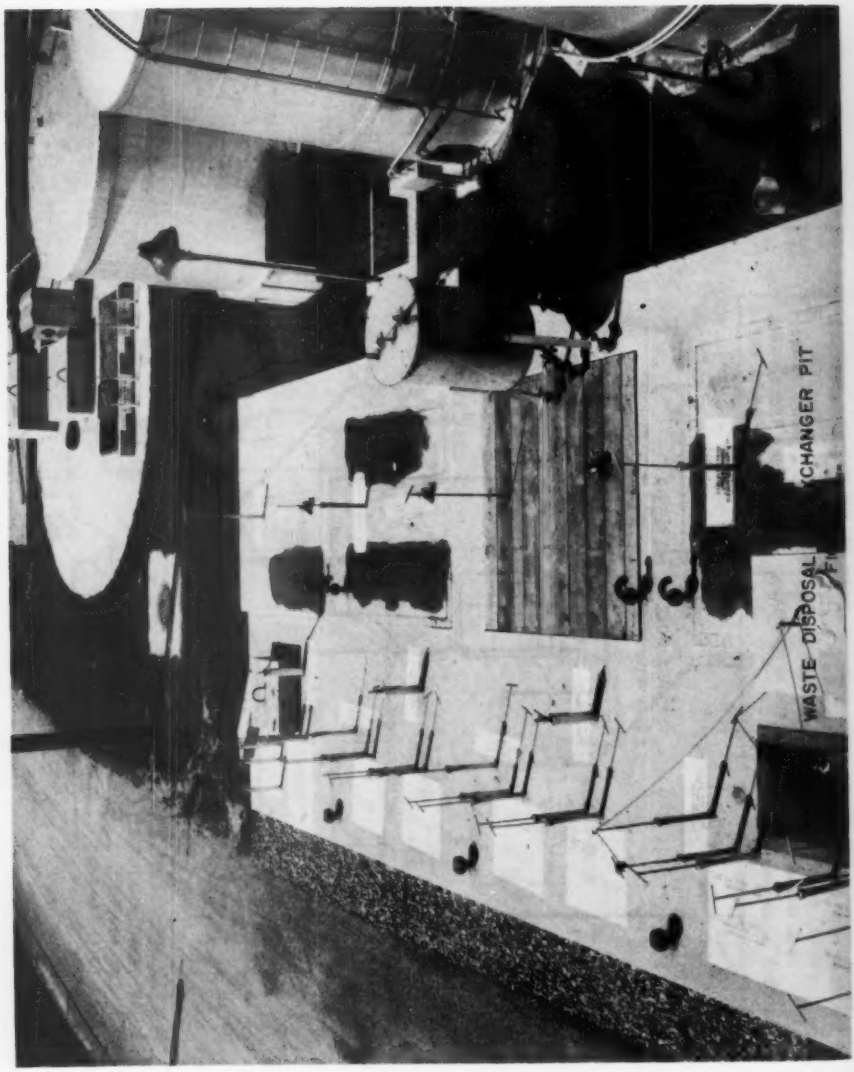
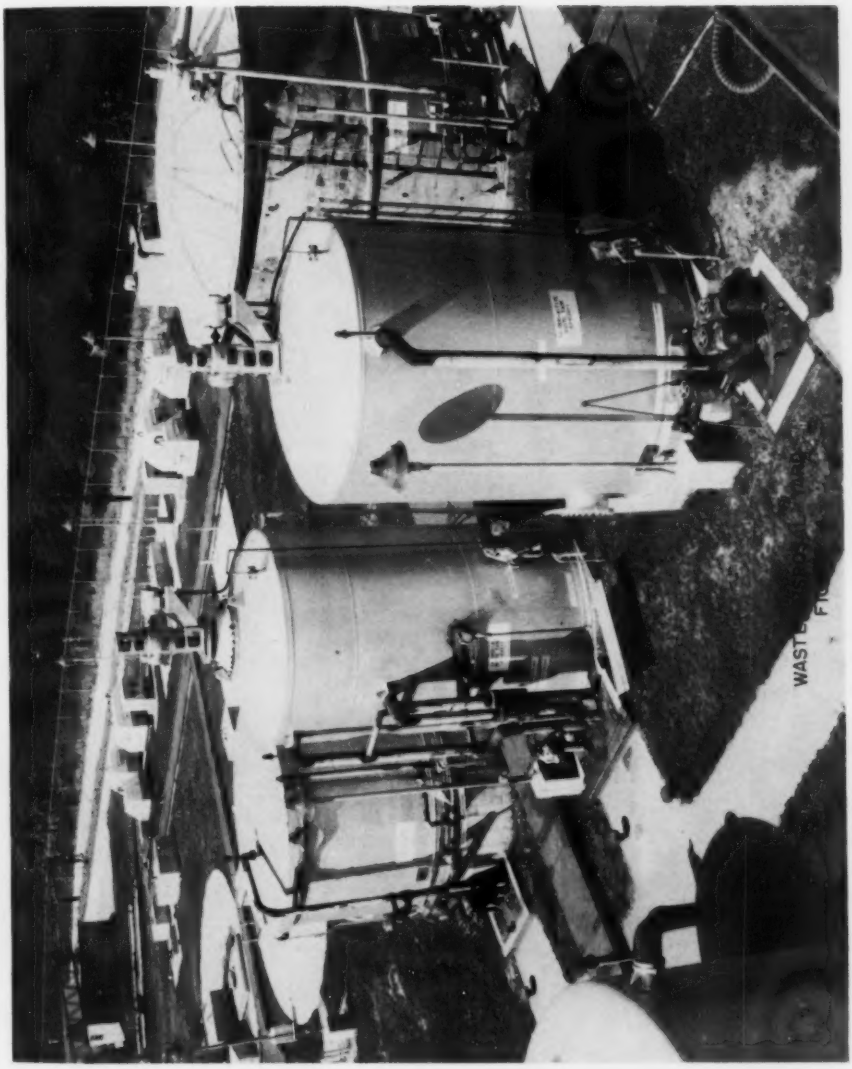
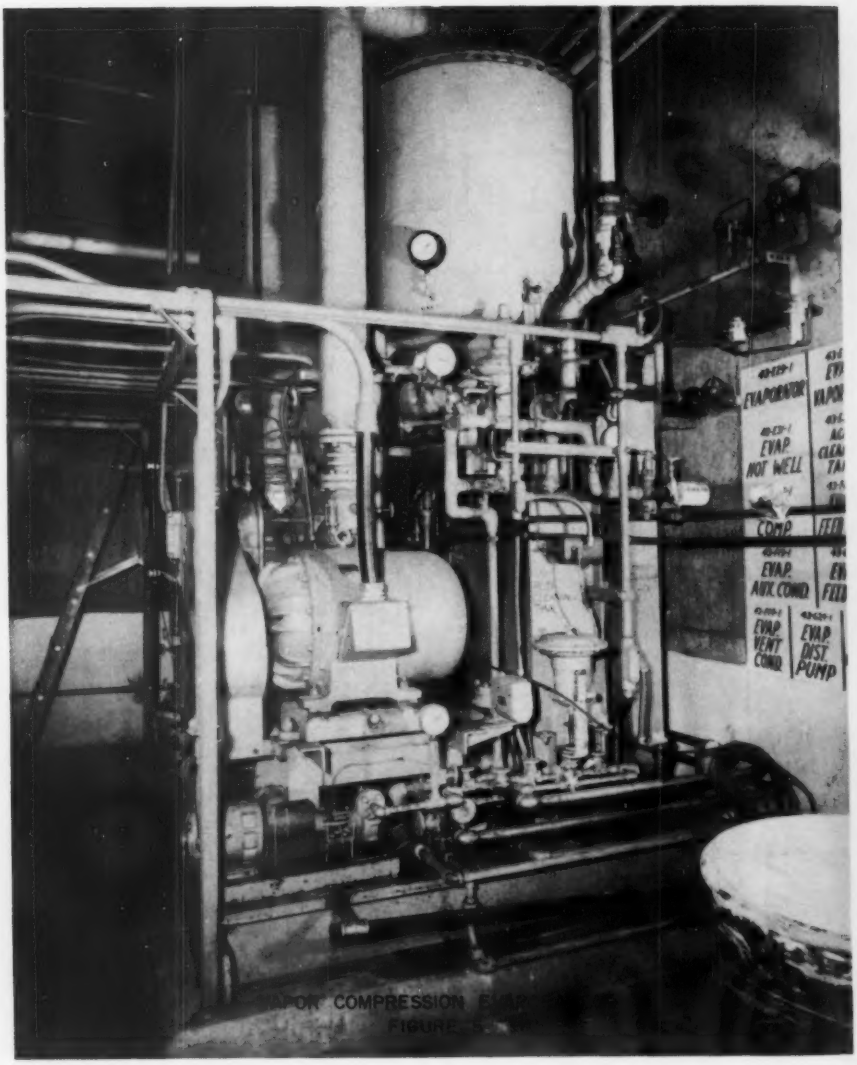


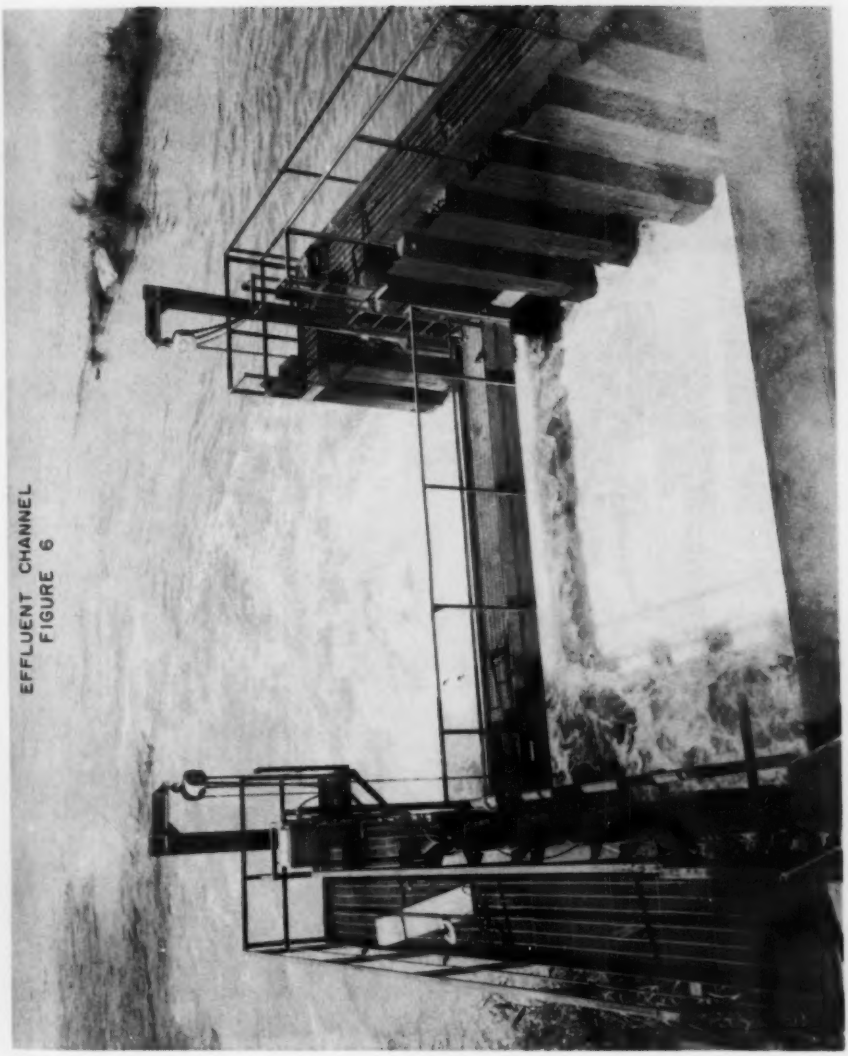
FIGURE 2
RADIOACTIVE WASTE DISPOSAL FACILITIES, SHIPPINGPORT ATOMIC POWER STATION


WASTE DISPOSAL
CHANGER PIT



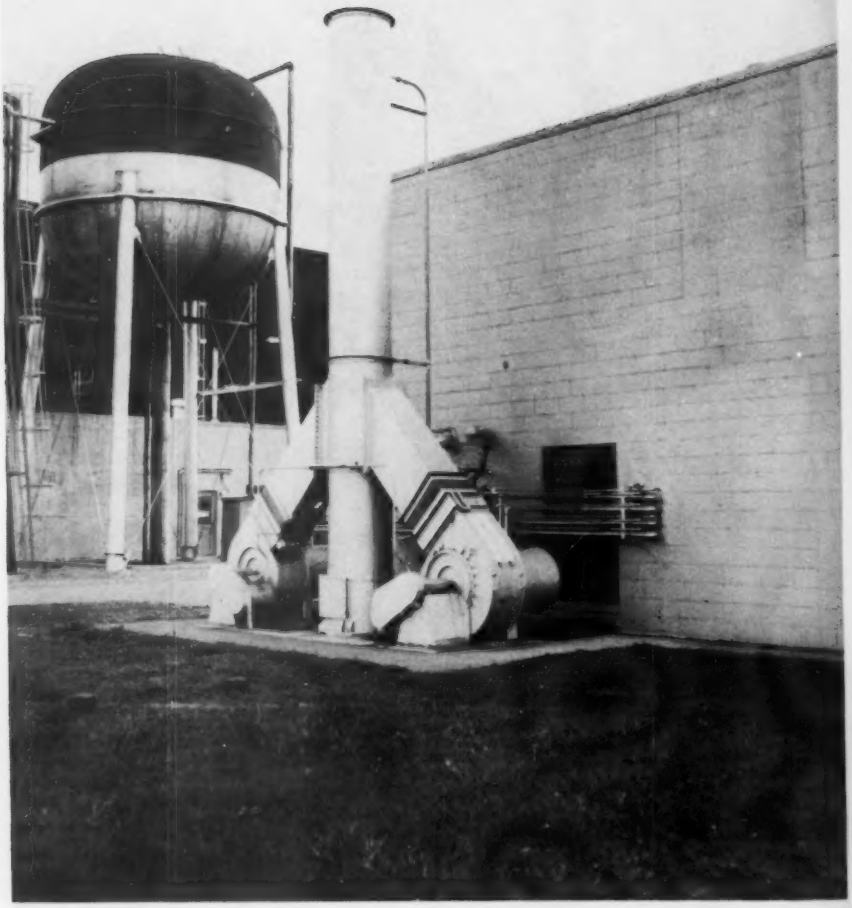


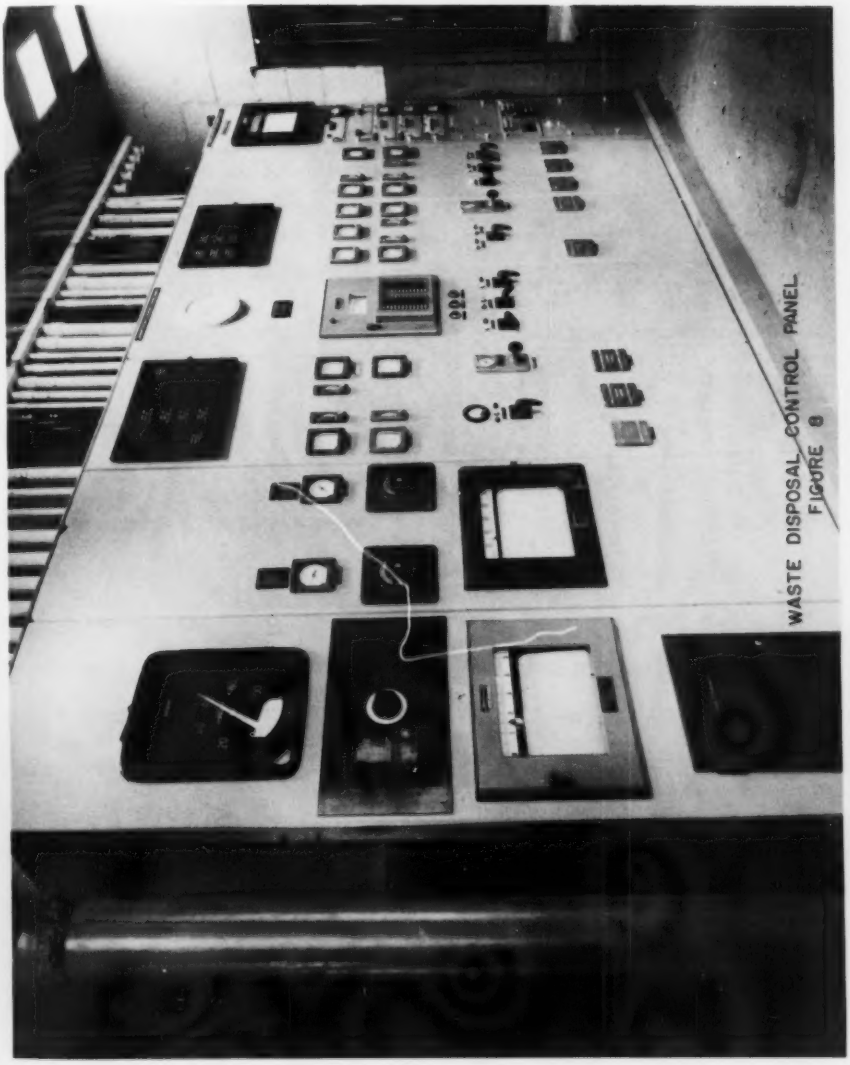
VAPOR COMPRESSION EVAPORATOR
FIGURE 5



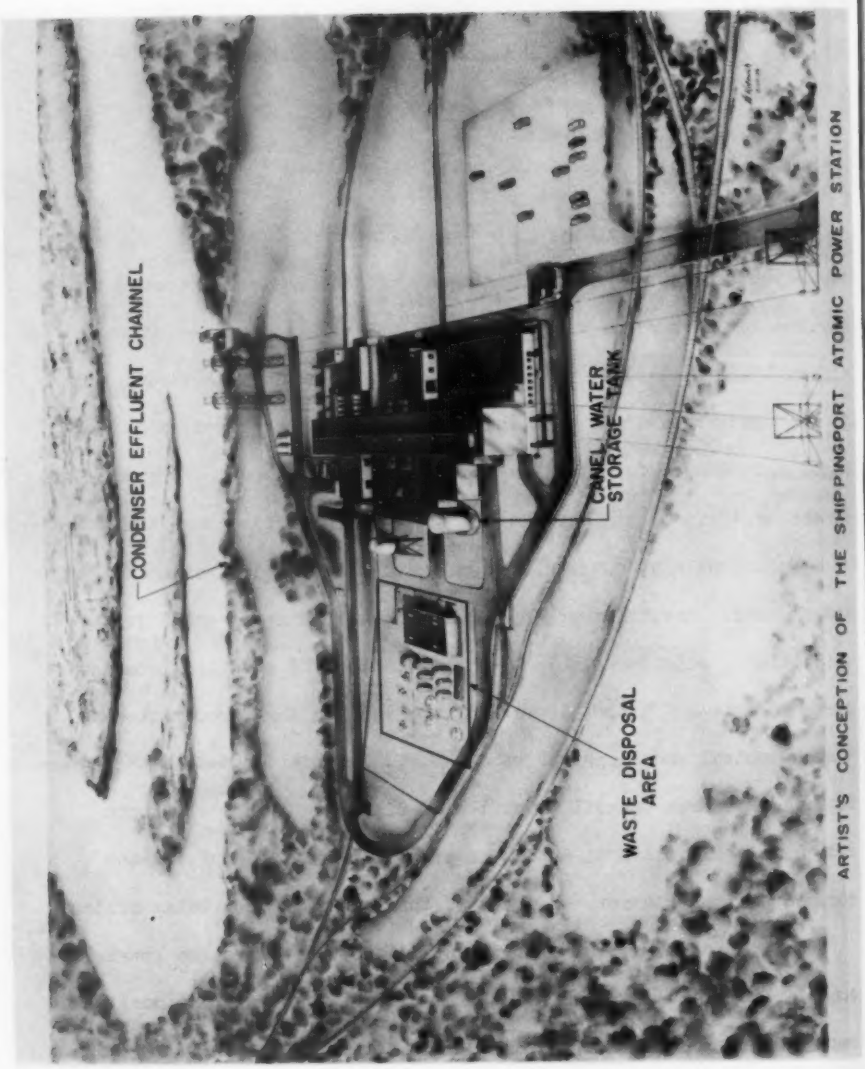
EFFLUENT CHANNEL
FIGURE 6

WASTE DISPOSAL STACK
FIGURE 7





WASTE DISPOSAL CONTROL PANEL
FIGURE 8



ARTIST'S CONCEPTION OF THE SHIPPINGPORT ATOMIC POWER STATION

Representative DURHAM. We have several statements on "Waste management operations" which I will place in the record at this point. (The statements referred to follow:)

The Role of Waste Management in the Development of the Nuclear Energy Industry

by

Joseph A. Lieberman, Sanitary Engineer
U. S. Atomic Energy Commission

Much has been written in a speculative and prognosticating way regarding the impact, or lack thereof, of radioactive wastes on the development of nuclear technology. Views have been expressed to the effect that this aspect of the industry is of negligible consequence to the opposite extreme that safe, adequate waste management is the major obstacle confronting the orderly, economic growth of application of the benefits of nuclear fission to medicine, agriculture and industry.

Unfortunately, outside of a relatively restricted scientific and technical treatment of the subject by a relatively small group of specialists, considerable lack of information and resulting confusion seems to exist. It is this situation which led the Subcommittee on Waste Disposal and Dispersal of the U. S. National Academy of Sciences-National Research Council Committee on Biologic Effects of Atomic Radiation to initiate an effort to prepare a summary, information document on the subject. A draft of this report is now being edited.

Outside of the specialist group radioactive wastes are generally considered as an uncategorized entity. Usually no distinction is made regarding the nature of the waste materials, its quantity, origin or the environment in which it must be considered. The word "radioactive" has been so strongly impressed in our minds that it has become the major all-inclusive descriptive term to the point where such important

ARTIST'S CONCEPTION OF THE SHIPPINGPORT ATOMIC POWER STATION

characteristics of wastes such as radioactivity concentration, total quantity of radioactivity, isotopic composition and chemical nature are often overlooked. Yet these are the characteristics which are the keys to waste management.

The purpose of this paper is to review and evaluate our past experience, both technical and administrative, in the waste management field and to use this in an effort to gain an insight into how this facet of nuclear technology might tie in with other considerations such as selection of sites for nuclear installations, the establishment of operating criteria for such installations and the administration of these criteria to assure protection of the public health and safety.

In any discussion of waste management it must be recognized that the primary management objective is the protection of man and his environment. Economic utilization of the "wastes" is obviously highly desirable but nevertheless secondary in comparison.

To further delineate the discussion it can be stated that there are three major, basic components involved as follows:

1. The maximum allowable quantity of specific isotopes in man or his various organs.

   This includes immediately the concept of maximum allowable concentrations of various isotopes in air and water, the ecological implications of biologic concentration of radioactivity by various organisms in our food chain and other highly important, complex and, in some instances, unknown biological considerations. From an engineering standpoint

the idea of some quantitative standard of permissible concentration of radioactivity in air and water is obviously important.

2. The specific nature of the radioactive waste under consideration.

This is a highly variable component and in order for proper consideration to be given to it, it must be approached in specific, quantitative terms. It must be completely understood that there is little basis for comparison of waste management techniques or problems associated with the liquid wastes emanating from a normally operating water cooled reactor, for example, and those associated with, let us say the aqueous re-processing of enriched uranium-aluminum alloy fuels.

3. The physical, chemical and biological characteristics of the environment in which the waste is to be considered or handled.

Included here, again in specific, quantitative terms (to the maximum extent possible); is knowledge or data of the atmosphere, the hydrosphere and the lithosphere relating to dilution and/or concentration of radioactivity in the environment.

Essentially, then, proper waste management consists of so identifying and quantitatively describing items 2 and 3, and <u>their combined behaviour</u>, so as to assure conformance to the standards

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established in item 1. Here a very important (but sometimes unrecognized) distinction is made between standards, and performance or operating criteria necessary to achieve these standards. For the most part, the standards are the result of the best available biological and medical knowledge and are of universal application. (One should, however, recognize the limitations of these standards due to lack of complete knowledge and, further, that such standards must be subject to modification as more and better knowledge is gained.) On the other hand because of the variability of the wastes and the environment, these components must be evaluated on an individual basis. The quantitative results of such an evaluation generally will not be of universal application.

Dispersal of Low Level Wastes

The two general philosophies of waste management have been noted previously and need not be dwelt on here. The "dilute and disperse" approach is applicable to wastes with generally low concentrations of radioactivity and to environments with proper dilution capacity. In other words, experience to date has shown that specific environments can be utilized to obtain a sufficient decontamination factor so that with the particular waste involved the standards of maximum allowable concentration in the air we breathe or the water we drink or the food we eat can be met. It is general practice to apply reduction factors of 10 to 100 to standards published by the International Committee on Radiation Protection and the U. S. National Committee on Radiation Protection and similar agencies in other countries in applying these standards to general populations.

In establishing performance criteria for waste dispersal operations from nuclear installations, two cardinal principles are therefore evolved.

1) The minimum practicable amount of radioactivity should be dispersed into the environment. "Practicable amounts" are determined by quantitative assessment of specific environments to assure protection of the public health and safety.

2) Continuing, periodic checking is mandatory to verify that performance criteria are being followed and accepted standards are being met. Performance criteria must be modified as changing requirements dictate.

If these principles are to be applied in a reasonable, meaningful way, it is apparent that designers and operators of nuclear facilities must have early knowledge of the quantity and characteristics of wastes to be managed and must further take this into proper consideration during the installation site selection process and the design and construction of the plant. Routine plant operation and various degrees and kinds of malfunction which may result in large "slugs" of wastes must receive equal attention. Adequate detention tank capacity to permit holdup of such slugs and treatment (if required) at efficient rates is generally required. Mechanical transport of large volumes of liquid wastes to dispersal points removed from the installation has been avoided in the United States burial grounds, for solid wastes have to date been centralized in relatively large non-populous, government owned reservations to the maximum extent possible.

In summarizing the situation with respect to dispersal operations in the United States, it can be said that our experience so far has demonstrated that safe, adequate systems for such operations can be established and necessary treatment or process components can be engineered provided sufficient basic information and data are made available. In the future as dispersal operations tend to expand and overlap and standards tend, perhaps, to become more stringent, it is probable that in-plant performance criteria will be receiving greater emphasis leading to increased treatment facilities and increased operational costs.

Disposal of High-Level Wastes

As distinguished from waste treatment, little actual work has yet been done on large scale ultimate disposal systems. At the present time it is our general conclusion that safe dispersal of large quantities of radioactivity in the environment is not feasible and future feasibility appears quite remote. Hence, our rather extensive tank storage program (capital cost estimated to date at about $\$65 \times 10^6$). It also seems fairly certain that interim tank storage will continue to be an integral part of any ultimate disposal scheme.

With regard to treatment of highly radioactive liquid wastes (primarily from reprocessing of irradiated fuels) for ultimate disposal, considerable investigation has been carried out on the fixation of radioactivity in inert, solid media. The objective is to so convert the wastes into a solid, non-leachable material that they be permanently stored (disposed) in specific environments with negligible long-term hazard.

The feasibility of direct disposal of high level wastes into certain selected geologic formations will only be determined after extensive laboratory and field experiments which are just now being initiated. Engineering and economic analyses are in their early preliminary stages, but sufficient work has been done to justify pursuit of such approaches at least on a limited scale. Of the geologic formations proposed in the United States, which include salt structures, deep synclinal basins, impermeable shales and certain porous formations, disposal into prepared cavities in salt appear to be most advanced.

Ocean disposal of high level wastes has been given some consideration. Quite often there has been confusion resulting from an inadequate categorization of quantity or nature of radioactive wastes involved. For high level fission product wastes emanating from chemical reprocessing, it appears conclusive that ocean disposal will not be feasible for many years to come if at all. Sufficient specific knowledge of oceanic behaviour is so lacking and attendant engineering problems appear so complex compared to other possible solutions to force the industry and government to systems that inherently have a greater degree of control. Again it is emphasized that this conclusion applies to the bulk high level wastes and not to smaller quantities of low and intermediate level wastes in both solid, packaged and liquid state.

Economic utilization of fission products would have important bearing on the high level waste problem, but at this writing the word "bearing" cannot be equated to "solution". Economic return from the

waste streams is obviously something to be sought but it must be remembered that from the standpoint of protection of public health and safety, the necessity of safe disposal systems will remain. In fact the problems of waste management may even be increased because of the wider use and dispersion of highly radioactive materials. In this connection it is also important to make the technical distinction between recovery of specific fission products for useful purposes and the essentially quantitative removal of these fission products as a waste disposal technic. Systems to accomplish the latter have yet to be developed.

Selection of sites which generate high level wastes is of utmost importance particularly with respect to ultimate long-term disposal. Many variables enter into this consideration. Availability of suitable disposal sites, the relative hazard and economics of transport of irradiated fuels and/or wastes, the types of fuel processed and the types of processing involved all play an important part. It appears quite likely, however, that installations generating high level wastes will be relatively few in number. This perhaps emphasizes the need for exceedingly careful analysis of siting. It must be pointed out, of course, that a single solution for all situations is neither required nor probably feasible. It seems quite certain that the industry will be required to give increased attention to an integrated evaluation of over-all systems of fuel re-processing, waste storage, waste treatment, and ultimate disposal. It is not unlikely that at least to some extent, reactor core design will also be included.

One of the most important questions entering into the role of waste management in the development of the nuclear industry is that of continuing long-term responsibility for radioactive wastes following actual disposal. Industry can and must assume increased responsibility for the physical operations involved in waste management. The performance criteria for these operations must be established co-operatively by industry and government and it appears that in meeting the long-term responsibility for protection of man and his environment, government must play the dominant part. It seems quite obvious that with respect to the latter, there are international as well as national implications.

Indeed, the industry-government relationship noted above may well be one of the major pillars to support the rational development of a nuclear industry that will be integrated fully and safely into our peacetime social and economic structure.

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INDUSTRIAL RADIOACTIVE WASTES: MINING AND MILLING OF URANIUM ORES

Grand Junction Operations Office, Grand Junction, Colorado

January 8, 1959

Introduction

The recent development of the uranium mining and milling industry has presented to this industry the new problem of the control of radioactive waste products. The waste products of uranium mines, similar to those of any mine, consist of ventilation air, waste rock, and water, if the mine is wet. Radioactivity of these mine wastes generally has been found to be of such a low order that no radioactive waste disposal problem is considered to exist. Within the mine proper the emanation of radon gas requires special considerations as discussed below.

The uranium ore processing mills are confronted with the problem of disposing of radioactive liquids and solid residues commonly referred to as tailings. The waste management operations of the uranium ore processing industry also will be discussed.

Mining

1. Physical Handling of Mine Wastes

A. Mine Air and Mine Ventilation

The major problem in uranium mine ventilation is the control of radon gas and radioactive dust. Radon gas emanates from uranium-bearing rock in proportion to the areas of such rock that are exposed by the mine workings and also in proportion to the amount of uranium and its disentegration products present in the rock. A

large, high grade uranium ore body will emanate radon at a greater rate than a smaller or lower grade ore body. Similarly, the rate of emanation will increase as the area of ore faces is increased.

The most practical method for removal of radon and its daughters from underground mine atmosphere is to move enough uncontaminated air into the various working places to sweep out the radon and to exhaust the resulting contaminated air to the outside. To accomplish this an adequate mine ventilation program is required. This is also true with regard to other air contaminants commonly found in uranium mines such as high silica-bearing dust and gases generated by the use of explosives and diesel powered equipment.

It is important to point out that all underground mines, regardless of the type ore produced, must provide sufficient ventilation to control toxic gases and harmful dust concentrations. State and Federal mining regulations specify the minimum amounts of ventilation to be provided. In field tests for determination of the degree of concentration of radon gas daughter products, it has been found that in many instances the elimination of hazardous quantities of silica dust and toxic gases has also substantially reduced the radioactive dust hazard.

The ventilation requirements in a mine are dependent upon such factors as the areal extent of the ore body; its continuity, thickness

and grade; the method of mining; i.e., room and pillar, long wall, etc; haulage layout; type of entry such as shaft, adit, or incline; depth of burial; number of underground employees; number of working faces needed to produce ore at the rate desired; and the rate of radon emanation. Minimum rule of thumb design specifications are: (1) Each man working in ore should be provided with 500 cubic feet of fresh air per minute from a tube located not more than 30 feet from the working face; (2) In all areas where radon is being emanated to the mine ventilation circuit, the rate of air flow should provide a complete change of air in the area at least once every four minutes.

The range of fresh air volume required may be between 1000 cu.ft./min. to over 200,000 cu.ft./min. dependent upon the size and character of the particular mine. Electrically driven blowers and fans capable of handling different volumes within this range are readily available from various manufacturers and they may be installed on the surface or underground as conditions dictate wherever electric power is available. Small mines remote from an electrical power service are restricted to the use of gas or diesel engine power and thereby to only surface fan installations, although small blowers powered by compressed air are available for auxiliary underground installation. Most of the larger uranium mines now have electric power.

1/ Woolrich, Paul F., Seven State Uranium Mining Conference on Health Hazards, February 22-23, 1955 - Utah State Industrial Commission

An example of good practice in the use of equipment to attain good ventilation would be about as follows:

1. A fan at the top or bottom of a smooth lined exhaust shaft having sufficient capacity to move the volume of air required for adequate ventilation of the mine down the intake shaft through the main air ways and up the exhaust shaft.

2. Auxiliary blowers along the fresh air airway to supply air to working faces. Such blowers can be moved from place to place when work in one place is finished and a new place is started or when a change allows shortening of a fan duct or tube. These ducts or tubes are usually made of impregnated fabric supported by a flexible wire frame. A 16 inch diameter tube will deliver about 2000 cu.ft./min. of air from a 5 HP high speed blower rated at 3000 cuft./min. at a distance of 400 to 500 feet from the blower, if, the tubing is maintained in good air-tight condition throughout its length and free of kinks or other obstructions.

3. Auxiliary blowers should be placed so that their intakes are in fresh air and away from contaminated return air from working faces. To do this requires careful planning of mine layout and constant supervisory attention to the coursing of both fresh and exhaust air currents. In this connection it should be pointed out that the pattern of mine workings is constantly

changing. New workings are continually being started and as workings advance they are often joined to other workings. Old workings are left behind and are no longer needed for airways nor is an air supply required for such abandoned areas. Fan tube discharge at working faces should be kept to within less than 30 feet of the working face to insure that the air current will reach it and sweep it.

4. Abandoned areas should be closed off from the passage of air currents by erection of suitable seals at the most advantageous locations. This will restrict the flow of fresh air to the areas where it is needed, and will prevent the radon emanations in abandoned areas from contaminating the air needed for ventilation of the active parts of the mine. Such seals can be made of concrete, gunited brattice, canvas brattice, or by walling off the opening with waste rock, provided, of course, they are maintained in air-tight condition.

5. Ventilation control doors should be built at locations where access is required between fresh airways and contaminated airways to prevent mingling of contaminated air with fresh air. Here again considerable attention and constant supervision are required to place and maintain proper air distribution. Such doors are also a good means of regulating volume of flow where splits off the main airways are required. Regulation is done

by installing stops to hold the particular door open the amount that will allow the required volume of air to pass.

B. Mine Water

In most uranium mines the problem of handling mine water is either non-existent because the mines are dry, or is minor with only intermittent pumping of small amounts of water from sumps or catch basins required. In the Ambrosia Lake, New Mexico area, however, pumping of mine water is a problem. Operating mines in that area are flowing as much as 600 gallons per minute. This amount of water is not excessive when compared to flows encountered in base metal and iron mining operations. Equipment to pump this amount of water is standard and readily available.

2. Mine Waste Disposal Costs

A. Cost of Ventilation

Combined capital and operating cost per ton of ore produced for uranium mine ventilation is in the range of 15 cents to $1.35. An average cost, as applied to all mines, is estimated to be 40 cents per ton. The data we have on which to base an estimate of this kind is incomplete and not too reliable. As the larger mines in Ambrosia Lake gain cost experience in ventilation, these figures may require some revision.

Although ventilation expenditures are part of mining cost, it should

be pointed out that in general the overall cost of producing a ton of ore under conditions of good ventilation is, in most instances, less than under conditions of poor ventilation. This is so because good ventilation adds to the efficiency of the underground labor force with a resulting reduction in labor cost.

B. Cost of Pumping

Combined capital and operating cost for pumping water from uranium mines ranges from nil to 75 cents per ton of ore produced. In the Ambrosia Lake area more experience will be needed before accurate costs are known; however, a reasonable estimate of the average cost of pumping is 25 cents per ton for all mines in the area.

Milling

1. Number of Mills, Range of Sizes, and Quantities of Material Handled

A. Ore Treated

As shown by Table I there are at present twenty-six uranium ore processing plants (including the two Union Carbide Nuclear Company concentrators) in operation in the western United States. The plants range in size from 50 to 3,500 tons of ore per day, and the total daily mill feed for all plants is about 21,000 tons.

The U_3O_8 content of the ores treated ranges from 0.10% up to about 1.0%, but is blended so that the daily average mill feed grade ranges from about 0.20% to 0.40% U_3O_8. For all mills, the average ore grade is approximately 0.25% U_3O_8.

B. Water Required

Fresh water required to treat uranium ores varies and depends upon the particular type of process employed. At present, the total water consumption of the industry amounts to about 12,500 GPM. Related to tonnage of ore treated, water consumption ranges from 0.2 to 1.4 GPM per ton of ore per day and averages about 0.6 GPM per ton of ore per day. This average stated in another way is equal to 3.6 tons, or 865 gallons, of water per ton of ore treated.

Such water requirements have influenced, and in some instances have dictated the specific locations of the uranium mills within certain mining areas. In many cases the mills have been located adjacent to major rivers from which process water is obtained. Other mills depend on wells that generally are drilled to depths of several hundred feet.

C. Concentrate Produced

The mills produce about 60 tons of uranium concentrates per day which contain between 70% and 92% U_3O_8. The concentrates, which are essentially impure uranium oxides and sodium and ammonium uranates, are delivered in steel drums to the Commission at Grand Junction, Colorado. There is very little radioactivity associated with the concentrates as the bulk of the highly radioactive decay products remains in the ore residue or tailings.

D. Tailings in Solid Form

Regardless of whether an alkaline or an acid leaching process is employed, there is no appreciable difference in the tonnage of solid tailings as compared with the dry tonnage of ore fed to process. The ore, however, is crushed and ground during processing so that the solid waste residue is essentially minus 28 mesh in size. These solids are mainly quartz sand grains with lesser amounts of calcite, gypsum, and clay minerals.

E. Tailings in Liquid Form

The quantity of tailings in liquid form is approximately equal to the quantity of fresh water used in the processing plants. Based on the industry average of 0.6 GPM of water per ton of ore treated per day, liquor in tailings would approach 865 gallons per ton of solids or about 18 million gallons per day.

F. Mill Ventilating Air

The quantity of mill ventilating air varies with the physical characteristics of the ore and the process employed. However, dust and fume collection facilities generally are installed in all areas within the mills where ore dust, concentrate dust, or chemical fume is encountered. Adequate dust control is essential in the ore crushing and sampling facilities, the sample preparation rooms, and the concentrate drying and packaging sections. Ventilation of the remainder of the mill is comparatively simple.

2. Physical Handling of Mill Wastes

Physically, the mill wastes are handled in the following manner:

Mill ventilating air is exhausted to the atmosphere. If dust-bearing, the ventilating air passes through dust collectors to remove the solids which are returned to process.

Generally, the solid and liquid tailings are combined and pumped to settling ponds near the mill site. Here the solids settle out and the excess water, less the amount lost by seepage, evaporation, and re-use, is allowed to overflow the dam. These settling ponds may be natural topographic basins or they may be man-made. In the case of the latter, if the terrain is suitable, a simple earthen dam across a small valley or draw may suffice; but at many of the milling sites it is necessary to construct ponds with dams surrounding the disposal area. These dams, or retaining walls, are constructed from materials scooped out of the pond area and may be only a few feet high at the start of operation. However, as the pond fills with solids, the heighth of the retaining walls must be raised. This is usually accomplished by using the sand portion of the tailings as a construction material.

At most of the mills there is no problem in impounding the solid tailings. However, disposal of the much larger volume of liquid tailings which are contaminated with soluble salts is a problem.

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This is particularly true at those mills located adjacent to streams or in populated areas where the local water supply is obtained from wells and is subject to contamination.

It has been mentioned that an average of 865 gallons of water are required in the processing of one ton of uranium ore. In the tailings pond, the amount of water retained by the solids will amount to 40 to 60 gallons per ton. The remainder less the amount lost by seepage and evaporation is disposed of by allowing the overflow to enter a river. If no suitable stream is available, the overflow percolates into the ground. At a few mills, where possible contamination of ground water supplies is a problem, attempts have been made or are being made to dispose of excess water by evaporation. This is accomplished by constructing large evaporation ponds with the bottom sealed by clay to prevent downward percolation.

The following table summarizes the method of disposing of liquid tailings at the twenty-six uranium processing mills now in operation in the United States:

Table A

Disposal of Liquid Tailings

Principal Method Employed	Number of Mills
By direct flow and/or seepage into a nearby stream	10
By evaporation and/or seepage to underground disposal	16

3. Capital and Operating Costs and Manpower Required for Mill Waste Disposal

The capital costs required for handling tailings vary considerably, depending upon the mill location, process employed, terrain features, soil permeability, etc. Generally, the initial capital cost to construct a dam complete with pipeline or launder system ranges from $50 to $100 per ton of ore per day. For example, the capital cost of a tailings dam for a 200 tons per day mill would range from $10,000 to $20,000.

Operating costs for tailings disposal also vary but, for an order of magnitude, given below are the Monticello tailings disposal costs for a representative period when both acid and alkaline leaching processes were in operation:

	$/Ton Ore Treated
Labor	0.18
Supplies	0.05
Utilities	0.04
Lime (for neutralization)	0.37
Total	0.64

Because of the mill location these costs are somewhat higher than the average for the industry. For those operations in isolated localities tailings disposal expense is considerably reduced.

The manpower required by the uranium milling industry as a whole for tailings disposal amounts to an estimated total of about 60 men. This labor requirement is not peculiar to uranium ore processing but is about the same as that required by other ore processing facilities, such as

copper or lead and zinc. No manpower requirement can be assigned specifically to mill ventilation air.

4. Current Efforts to Control the Disposition of Radioactive Mill Wastes

Until just recently very little attention has been given to the problem of radioactive waste control throughout the domestic uranium-ore processing industry. Even now it appears that the only waste constituting a potential radioactive hazard is the tailings pond overflow or seepage that might possiboy contaminate ground waters or streams. Where hazards have been found to exist, provisions have been made to prevent or to limit to a predetermined quantity the overflow from the tailings area.

The mill management, through an initial survey by the Division of Inspection, has been acquainted with the provisions of CFR-10, Part 20 - Standards for Protection Against Radiation. Through private radioactivity surveys, each mill will be required to appraise its own particular effluent problems and take whatever steps are necessary to be in compliance with the permissible levels of radiation as set forth in Part 20.

The AEC is actively studying the problem of radioactive wastes and is attempting to develop economical means of reducing radiation hazards. This program is being conducted by the Raw Materials Development Laboratory at Winchester, Massachusetts, and the results of the laboratory studies are being evaluated under actual operating conditions at the

TABLE I - URANIUM ORE PROCESSING PLANTS - GENERAL INFORMATION ON SIZE, PROCESSES AND TAILINGS

	Company	Location	Capacity in Tons of Ore Per Day	Type of Process
1.	Union Carbide Nuclear Company	Uravan, Colorado	1,000	Acid Leach, CCD,
2.	Union Carbide Nuclear Company	Rifle, Colorado	400 1/	Acid Leach, CCD,
3.	Union Carbide Nuclear Company	Greenriver, Utah	300	Sand-Slime Separa Chemical Precipi
4.	Union Carbide Nuclear Company	Slick Rock, Colorado	300	Sand-Slime Separa Chemical Precipi
5.	Climax Uranium Company	Grand Junction, Colorado	330	Acid Leach, CCD,
6.	Vanadium Corporation of Amer.	Durango, Colorado	750	Acid Leach, Alkal Precipitation, R
7.	Trace Elements Corporation (UCN)	Maybell, Colorado	300	Acid Leach, RIP
8.	U.S. Atomic Energy Commission	Monticello, Utah	350	Alkaline Leach -
9.	Vitro Uranium Company	Salt Lake City, Utah	600	Acid Leach, SX
10.	Uranium Reduction Company	Moab, Utah	1,500	Acid Leach, RIP
11.	Texas-Zinc Minerals Corporation	Mexican Hat, Utah	1,000	Acid Leach, SX
12.	Kerr-McGee Oil Industries, Inc.	Shiprock, New Mexico	300	Acid Leach, SX
13.	The Anaconda Company	Bluewater, New Mexico	3,500	Two Plants, Alkal cipitation and A
14.	Mines Development, Inc.	Edgemont, South Dakota	400	Acid Leach, RIP -
15.	Rare Metals Corp. of America	Tuba City, Arizona	300	Acid Leach, RIP
16.	Western Nuclear Corporation	Split Rock, Wyoming	400	Acid Leach, RIP
17.	Dawn Mining Company	Ford, Washington	400	Acid Leach, CCD,
18.	Lucky Mc Uranium Corporation	Gas Hills, Wyoming	750	Acid Leach, CCD,
19.	Homestake-New Mexico Partners	Grants, New Mexico	750	Alkaline Leach Ca
20.	Gunnison Mining Company	Gunnison, Colorado	200	Acid Leach, SX
21.	Cotter Corporation	Canon City, Colorado	50	Alkaline Leach -
22.	Fremont Minerals, Inc.	Riverton, Wyoming	500	Two Circuits, one Alkaline Leach
23.	Homestake-Sapin Partners	Grants, New Mexico	1,500	Alkaline Leach Ca
24.	Phillips Petroleum Company	Grants, New Mexico	1,725	Alkaline Leach Ca
25.	Kermac Nuclear Fuels Corp.	Grants, New Mexico	3,300	Acid Leach - SX
26.	Lakeview Mining Company	Lakeview, Oregon	210	Acid Leach - SX
	Total		21,115	

1/ Feed also consists of upgraded material from Greenriver and Slick Rock concentrators.

2/ Abbreviations used:

CCD is countercurrent decantation

SX is solvent or liquid-liquid extraction

RIP is resin-in-pulp

Grand Junction Operations Office
January 8, 1959

AND TAILINGS DISPOSAL

e of Process Employed 2/	Method of Tailings Disposal
Leach, CCD, Ion Exchange	Sands are stacked by conveyor, slime and waste liquor pumped to ponds.
Leach, CCD, Solvent Extraction (SX)	Sands, Slimes and waste liquor combined and pumped to pond.
Slime Separation, Acid Leach of Sands, ical Precipitation from Acid Liquor	Sands and barren solution pumped to pond.
Slime Separation, Acid Leach of Sands, ical Precipitation from Acid Liquor	Sands and barren solution flow to pond.
Leach, CCD, SX	Sands and barren solution pumped to pond. Slimes pumped separately to same pond.
Leach, Alkaline Leach - SX, Chemical ipitation, Reduction Fusion	All solids and solution pumped to pond which overflows to river.
Leach, RIP	Sands, slimes and waste liquor all pumped to pond.
ine Leach - RIP	All tailings pumped to pond.
Leach, SX	Solids with some liquor pumped to pond, remaining liquor pumped to separate pond.
Leach, RIP	All tailings combined, partially neutralized and pumped to pond.
Leach, SX	All tailings combined, neutralized, and pumped to pond.
Leach, SX	Solids with some liquor pumped to pond, remaining liquor pumped to separate pond.
lants, Alkaline Leach - Caustic Pre-tation and Acid Leach RIP	All tailings combined, neutralized, and pumped to pond.
Leach, RIP - SX	Sands impounded separately, slimes and waste liquor put in pond.
Leach, RIP	All tailings combined and pumped to pond.
Leach, RIP	All tailings combined and pumped to pond.
Leach, CCD, Ion Exchange	All tailings combined and pumped to pond.
Leach, CCD, Ion Exchange	All tailings combined and pumped to pond.
ine Leach Caustic Precipitation	All tailings combined and pumped to pond.
Leach, SX	All tailings combined and pumped to pond.
ine Leach - Caustic Precipitation	All tailings combined and pumped to pond.
ircuits, one Acid SX, the other line Leach	All tailings combined, neutralized, and pumped to pond.
ine Leach Caustic Precipitation	All tailings combined and pumped to pond.
ine Leach Caustic Precipitation	All tailings combined and pumped to pond.
Leach - SX	All tailings combined and pumped to pond.
Leach - SX	All tailings combined and pumped to pond.

rs.

Tailings Pond locations To Towns and Rivers

Plant is on San Miguel River, Tailings area is on hill 1 mile from river. Pond overflows 50 GPM to river.

2 miles from Rifle, some effluent to Colorado River

2 miles from Green River, no effluent to river.

Plant is ¼ mile from Dolores River, tailings area on river bank. Seepage but no direct overflow to river.

Edge of Grand Junction city limits, some effluent to Colorado River nearby.

Plant and tailing area on Animas River bank in Durango. Pond overflow to river.

Town of Maybell and Yampa River 5 miles away. Pond effluent flows to intermittent creek thence to river.

Edge of Monticello city limits - tails impounded, effluent overflows to South Creek.

Edge of Salt Lake City - some effluent to Jordan River 1 mile away.

On Colorado River, 1 mile from Moab, Utah - effluent to river

On Navajo Reservation not near any town - San Juan River 1½ miles away receives some effluent.

Very near San Juan River, 1 mile from Shiprock, pond liquor evaporates and is lost by seepage.

No rivers in area, 1 mile from small settlement - pond liquor evaporates and is lost by seepage.

Tailing pond in Edgemont and near Cheyenne River. No overflow to river. No known seepage to river.

On Navajo Reservation 6 miles from Tuba City and away from any rivers. Pond effluent lost by seepage and evaporation.

Company settlement 2 miles from mill - creek 1½ miles away. No pond effluent.

Not near any town or river - no pond effluent.

No town or river nearby. Very isolated area.

No river in area, 6 miles from Grants. Pond liquor evaporates or is lost by seepage.

Plant and tailings pond 2 miles from town of Gunnison. Tailings ½ mile from Gunnison River. No pond overflow. Pond liquor lost by seepage.
No river in area. Pond liquor lost by seepage.

Plant and tailings pond adjacent to the Wind River. Seepage but no direct pond overflow to river.

No river in area, 6 miles from Grants. Pond liquor evaporates or is lost by seepage.

No river in area, 23 miles from Grants. Pond liquor evaporates or is lost by seepage.

No river in area, 23 miles from Grants. Pond liquor evaporates or is lost by seepage.

No river in area. Plant located at edge of town.

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Commission-owned mill at Monticello, Utah. Topical Report WIN 114, "Survey and Prevention Techniques for Control of Radioactivity Hazards at the Monticello Mill" describes our activities in this connection. The results of the studies and tests are being made available to mill management. At the same time, some of the larger and better equipped milling companies are doing independent research along these lines.

Attachment:
Table I - Uranium Ore Processing Plants - General Information on Size, Processes and Tailings Disposal

WASTE MANAGEMENT OPERATIONS AT THE FEED MATERIALS PRODUCTION CENTER

Fernald, Ohio

Prepared by R. C. Heatherton and K. N. Ross, Health and Safety Division
National Lead Company of Ohio

February, 1959

Biographical Sketch - R. C. Heatherton

Name: Richard C. Heatherton

Date & Place of Birth: April 2, 1919, Decatur, Illinois

Education: B.S. in Chemistry, 1943, University of Illinois, Urbana, Illinois

Employment History:

May 1943 - March 1946	Linde Air Products Company Tonawanda, New York Ceramics Plant, Manhattan District Project Chemist
March 1946 - Sept. 1947	Victor Chemical Company Chicago Heights, Illinois Chemical Engineer, Semi Works
Sept. 1947 - July 1949	Linde Air Products Company Ceramics Plant - AEC Operations Health Physicist
Aug. 1949 - Sept. 1951	USAEC, New York Operations Office Health & Safety Division Industrial Hygienist
Sept. 1951 - Oct. 1957	National Lead Company of Ohio Health & Safety Division Head of Industrial Hygiene and Radiation Department
Nov. 1957 - present	National Lead Company of Ohio Health & Safety Division Head of Analytical Laboratory Department

Biographical Sketch - K. N. Ross

Name: Keith N. Ross

Date & Place of Birth: March 11, 1927, Wheeling, West Virginia

Education: B.S. in Chemistry, 1951, West Liberty State College
West Liberty, West Virginia

Employment History:

Nov. 1951 - May 1953	Brush Beryllium Company Luckey, Ohio Laboratory Technician
May 1953 - Dec. 1953	Brush-Be Copper Division Industrial Hygienist
Jan. 1954 - Oct. 1956	National Lead Company of Ohio Technical Division Analytical Chemist
Nov. 1956 - present	National Lead Company of Ohio Health & Safety Division Industrial Hygienist

Introduction

A feed materials production center such as the one at Fernald, Ohio, combines the various metallurgic processes essential to the production of reactor fuel elements from uranium ores and mill concentrates. These processes include: acid digestion of the ores and concentrates, organic phase extraction of uranyl nitrate, subsequent conversion of the nitrate to uranium oxides and tetrafluoride, reduction to uranium metal, and fabrication of the metal into fuel elements. The Fernald operations also include plants for sampling of the ores and concentrates and for recovery of uranium from various residues.

Both liquid and solid waste materials are generated in the described processes. These wastes contain varying quantities of naturally-occurring radioisotopes, principally those which occur in the uranium series.

The management of these wastes is governed by several factors. Refining and fabrication wastes are considerably less, both in quantity and activity level, than those produced by a nuclear reactor. Nevertheless, the long half-lives of some of the isotopes and their known or estimated radiological effects make control of their storage or of disposal necessary. In addition to their radioactivity, some of the wastes contain other noxious materials, for example, fluoride and nitrate. Some solid wastes have at least potential value as a source of additional uranium and other valuable metals. Finally, surface and ground water characteristics in the Fernald area require control measures which would not be necessary in other areas.

Hydrologic Factors

The Fernald plant is located on a 1200-acre tract in the Great Miami River valley. This river, which receives the plant's liquid waste, has its origin about 170 miles to the northeast. The watershed which contributes to its flow encompasses some 5300 square miles.(1) The river flow normally varies during the year from a low of about 300 second-feet to a high of over 30,000 second-feet. Usually there is one or more periods of low flow sustained for several days in the fall of the year. It is during these periods of low flow that the liquid waste disposal problem becomes acute.

The nearest approach of the river to the plant is about one mile to the east. From this point the river flows southwest until it empties into the Ohio River some twenty miles beyond the plant.

To the west of the plant flows another small stream called Paddy's Run. At times there is virtually no flow in this stream. Following heavy rainfall, the flow may amount to a few hundred

second-feet. This stream receives only some surface runoff from the plant during rainstorms.

While the Miami River is a source of water supply for some upstream communities, downstream use is limited to industrial purposes by one or two plants. Paddy's Run is used to a minor extent for watering stock.

The main source of water in the Fernald area is an aquifer nearly 200 feet in depth. Flow in the aquifer is unconfined and in the general direction of the Miami River.(2) This aquifer is the source of water used by the plant. It also supplies several million gallons of water each day to other wells, including two wells of a commercial supplier.

Waste Treatment Facilities and Practices

Original Design

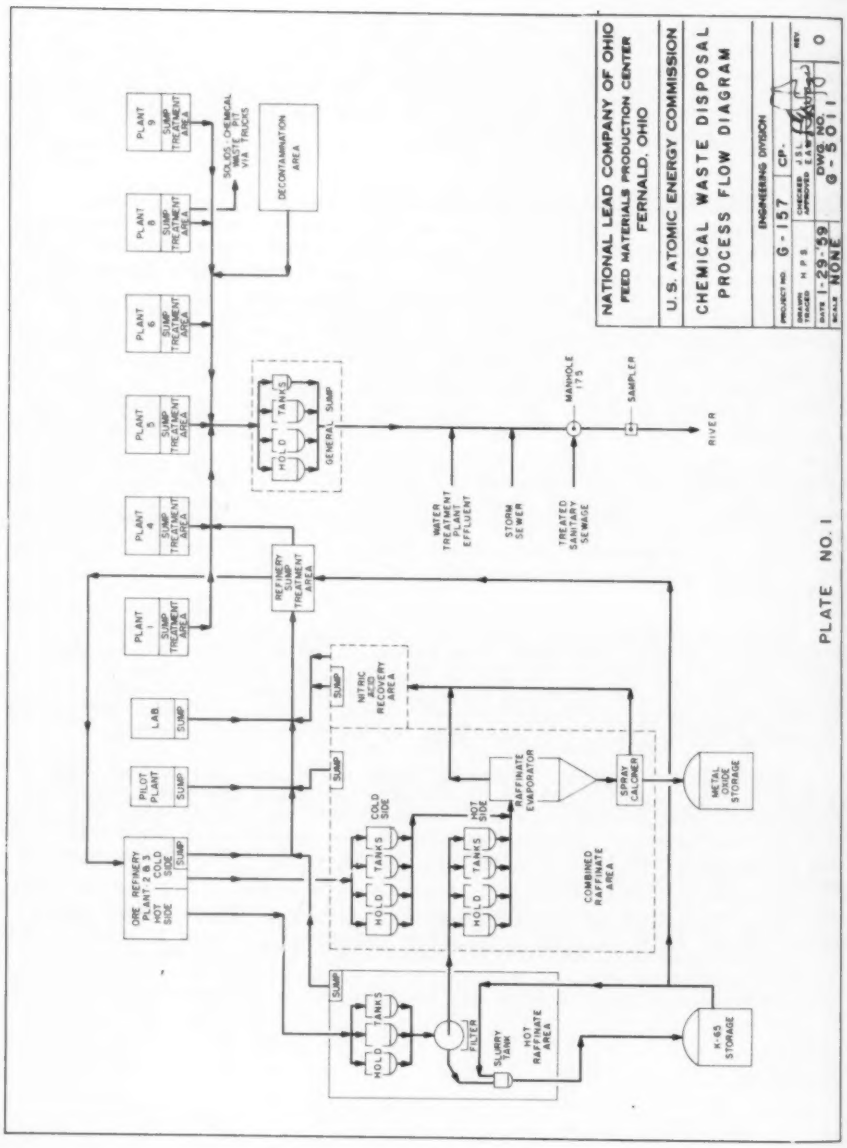
The process flow for wastes as originally designed is shown in Plate 1. The basic design included collection sumps and treatment equipment for each plant. The purpose of these facilities was the removal of uranium from waste water for economic reasons. In its simplest form the treatment facility was a hold tank for sump liquors and a filter for removal of suspended solids. Other plants required to treat acid solutions had sufficient tankage for neutralization and precipitation of uranium preceding filtration.

The extraction process in the ore refinery is the major source of radioactive waste. Special treatment facilities for these wastes were provided. These facilities included, in addition to the necessary hold tanks for raffinate, filters for removal of nitric insoluble material, an evaporator for concentration of the filtered raffinate, and a spray calciner for conversion of the residual nitrates to metal oxides.

Only raffinates from digestion of pitchblende ores were filtered to remove the insoluble material before evaporation and denitration. Contained in the filter cake was radium from the original ore. Substantially complete radium removal was achieved by barium sulfate co-precipitation. When necessary, prior additions of barium and sulfate ion were made to achieve this removal.

Prestressed concrete storage tanks were provided for storage of the radium cake. Similar tanks were provided for metal oxides. Transfer of the radium cake was accomplished by reslurrying and pumping the slurry. Neutralization of the slurry with soda ash was first achieved before pumping. Metal oxides were pneumatically conveyed to storage tanks.

Also included in the original design was a general sump which was to serve as a central collection point for process liquid wastes


NATIONAL LEAD COMPANY OF OHIO
FEED MATERIALS PRODUCTION CENTER
FERNALD, OHIO
U.S. ATOMIC ENERGY COMMISSION
CHEMICAL WASTE DISPOSAL
PROCESS FLOW DIAGRAM
ENGINEERING DIVISION
PROJECT NO. G-157 CP-
DRAWN H.P.S.
CHECKED J.S.L.
APPROVED E.A.W.
DATE 1-29-59
SCALE NONE
DWG. NO. G-25011
REV. 0
PLANT 9 SUMP TREATMENT AREA
PLANT 8 SUMP TREATMENT AREA
PLANT 6 SUMP TREATMENT AREA
PLANT 5 SUMP TREATMENT AREA
PLANT 4 SUMP TREATMENT AREA
PLANT SUMP TREATMENT AREA
LAB. SUMP
PILOT PLANT SUMP
ORE, REFINERY PLANT 2 & 3 COLD SIDE SUMP
HOT SIDE
SOLIDS - CHEMICAL WASTE PIT VIA TRUCK
DECONTAMINATION AREA
HOLD TANKS
GENERAL SUMP
REFINERY SUMP TREATMENT AREA
NITRIC ACID RECOVERY AREA
SUMP
COLD SIDE
HOT SIDE
RAFFINATE EVAPORATOR
SPRAY CALCINER
METAL OXIDE STORAGE
COMBINED RAFFINATE AREA
SLURRY TANK
FILTER
HOT RAFFINATE AREA
K-65 STORAGE
WATER TREATMENT PLANT EFFLUENT
STORM SEWER
TREATED SANITARY SEWAGE
MANHOLE 175
SAMPLER
RIVER
PLATE NO. I

to be discharged to the river. General sump tanks provided were four in number, having a total capacity of 65,000 gallons. No further treatment of sump liquors was provided for the general sump.

Expansion Facilities

The treatment facilities for the individual plants have been adequate. These systems have operated as planned with only minor changes in face of increasing plant production. Only the raffinate systems and general sump were found to be inadequate.

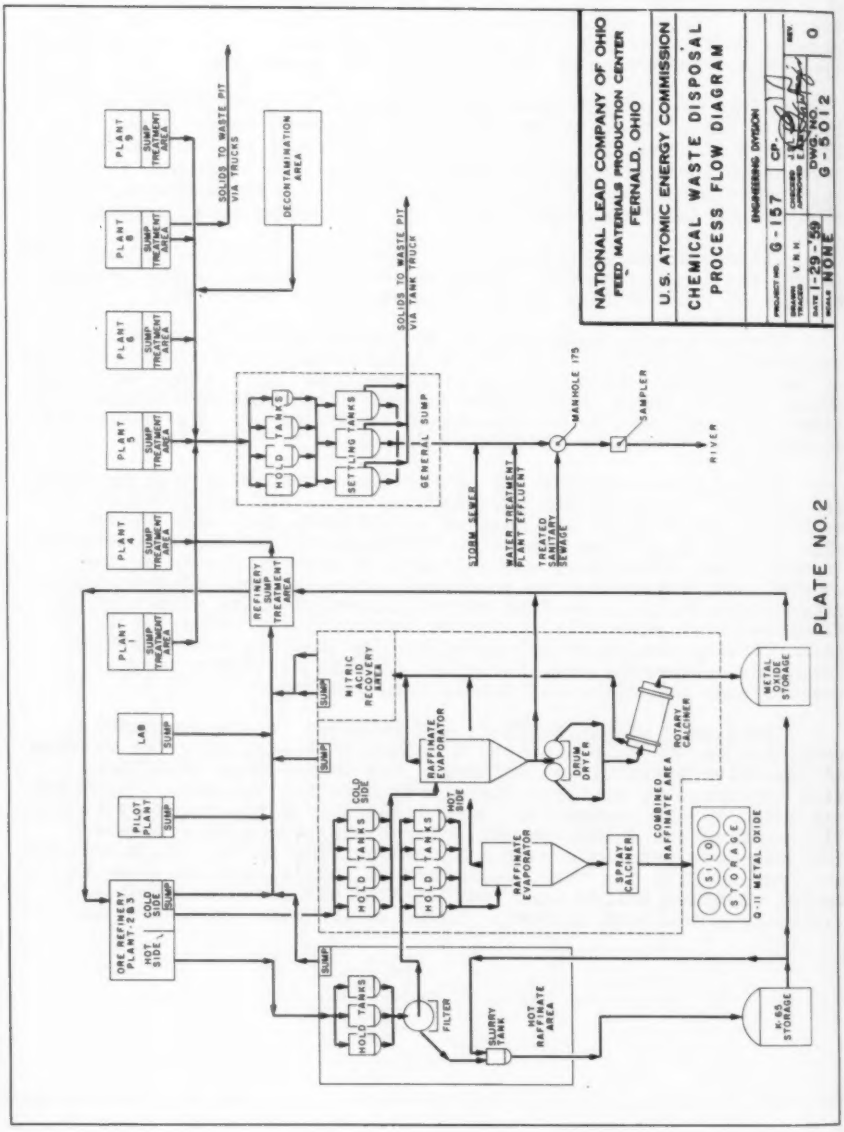
As the refinery production increased, the spray calciner became a bottle-neck in the raffinate treatment operations. Operating difficulties inherent in the calciner and corrosion problems requiring excessive maintenance to the equipment added to the problem. An additional requirement was added for separation of metal oxides from pitchblende ores from other metal oxides.

Silos originally intended for storage of ores and concentrates at the sampling plant were converted to pitchblende oxide storage. Storage of other oxides in the concrete tanks continued.

During a major expansion of the plant in 1955 a second evaporator, drum dryer and rotary calciner were added for treatment of cold raffinate from uranium concentrates. Also added were three 50,000 gallon tanks at the general sump to provide additional hold capacity for waters to be discharged to the river. The additions to the waste treatment facilities are shown in Plate 2.

Although the denitration of raffinate was originally included for economic reasons, for the recovery of nitric acid, and as a vendor requirement in the case of pitchblende ore, this system also served to prevent excessive additions of radioactive material to the river. Periodic, small losses of some raffinate to the general sump resulted from equipment failure. Some of the raffinates--especially those from recycled uranium which were high in fluoride--were extremely corrosive and costly to process. Their radioactivity consisted mainly of the UX_1 and UX_2 beta-emitting daughters. Nevertheless, diversion of this raffinate to the general sump and river was deemed inadvisable.

Since most of the activity was attached to the suspended particles in the sump liquors, a method of alleviating the release of excessive activity was offered. The three large hold tanks were converted to settling tanks with floating decants. "Separan" was added to the neutralized sump liquors from the refinery to aid flocculation and solid removal. The tanks were operated on a fill-and-draw principle over a four-hour time cycle. This included one hour to fill the tank, two hours to settle, and one hour to empty. Settled solids were periodically withdrawn as a thick slurry from the tank bottoms. These solids were trucked to the


NATIONAL LEAD COMPANY OF OHIO
FEED MATERIALS PRODUCTION CENTER
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U.S. ATOMIC ENERGY COMMISSION
CHEMICAL WASTE DISPOSAL
PROCESS FLOW DIAGRAM
ENGINEERING DIVISION
PROJECT NO. G-157
DRAWN V.H.H.
DATE 1-29-59
SCALE NONE
DWG. NO. G-5012
REV. 0
PLATE NO. 2
ORE REFINERY PLANT 2&3 SIDE
PILOT PLANT
LAB
PLANT 1
PLANT 4
PLANT 5
PLANT 6
PLANT 8
PLANT 9
SUMP TREATMENT AREA
REFINERY SUMP TREATMENT AREA
SOLIDS TO WASTE PIT VIA TRUCKS
DECONTAMINATION AREA
SOLIDS TO WASTE PIT VIA TANK TRUCK
HOLD TANKS
SETTLING TANKS
GENERAL SUMP
STORM SEWER
WATER TREATMENT PLANT EFFLUENT
TREATED SANITARY SEWAGE
MANHOLE 175
SAMPLER
RIVER
NITRIC ACID RECOVERY AREA
RAFFINATE EVAPORATOR
DRUM DRYER
ROTARY CALCINER
METAL OXIDE STORAGE
COLD SIDE
HOT SIDE
SPRAY CALCINER
COMBINED RAFFINATE AREA
G-11 METAL OXIDE STORAGE
FILTER
SLURRY TANK
HOT RAFFINATE AREA
K65 STORAGE

chemical waste pit for storage. Careful operation of this system has permitted continued operation of the refinery with a plant effluent suitable for release to the river under exacting control requirements.

During the last year a complete study was made of the raffinate waste treatment. When the receipt of pitchblende ores ceased, there was no longer any necessity to store radium cake or metal oxides for the vendor. The cost of operating and maintaining the calciners have reached the point of uneconomical return.

A final alteration to the system for collection and treatment of all waste waters is nearing completion. A new storage pit of 226,000 cubic yards capacity was added. This pit will serve as a settling basin for removal of settleable solids from the waste. It will also provide storage for these solids. Neutralized raffinate will be pumped directly to the pit with the expectation that the water which is decanted to the clear-well will be virtually free of solids and radioactivity. The latest revision to the system and flow of waste is shown in Plate 3.

Storage of Solids

The first chemical waste pit was provided in the original design primarily for storage of the waste magnesium fluoride from the scrap recovery plant. This pit was dug into clay and was clay-lined on the sides and bottom to minimize seepage of contaminated water into the ground. Design details for the pit have been described in another publication.(3)

Use of decant lines provided for release of water from the pit was not found to be necessary since the rate of evaporation was sufficient to keep the water volume in check.

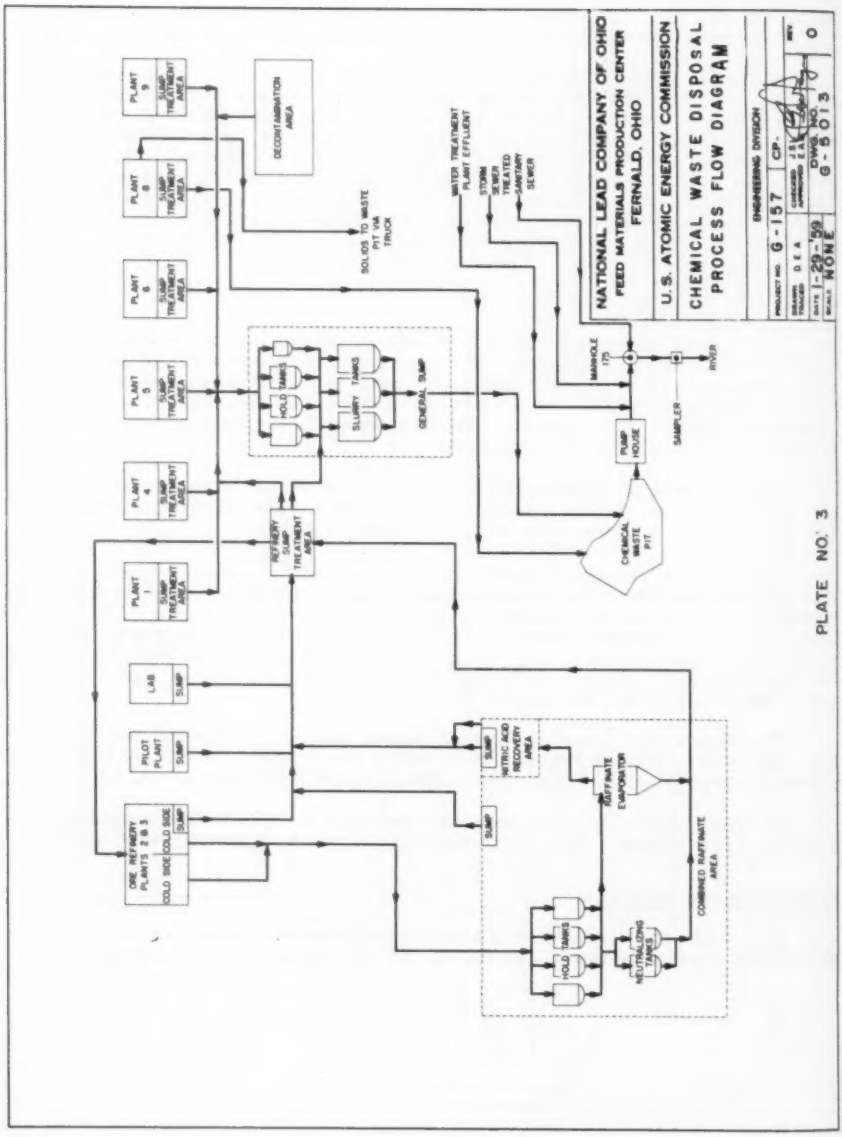
The second chemical waste pit was added to receive the slurried waste from the uranium ammonium phosphate recovery process. Construction features resembled those of the first waste pit. Because of the larger water additions from both the recovery plant slurry and the general sump tank bottoms, it was necessary to provide for drawing off water. This water was routed by way of the industrial waste water line to the Miami River.

The third pit which has been added was constructed, as were the other two, by excavation into clay and lining the interior to prevent seepage of contaminated liquids into the water table.

Plate 4 shows the lay-out for the pits. These are remotely located with respect to the production area and plant water supply.

Miscellaneous Wastes

Although the accumulation of miscellaneous type waste materials is a continuing problem, it has always been possible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U.S. ATOMIC ENERGY COMMISSION
CHEMICAL WASTE DISPOSAL
PROCESS FLOW DIAGRAM
ENGINEERING DIVISION
PROJECT NO. G-157 CP-
DRAWN TRACED D.E.A.
CHECKED J.E.K.
DATE 1-29-59
SCALE NONE
DWG. NO. G-5013
REV. 0
PLANT 9 SUMP TREATMENT AREA
PLANT 8 SUMP TREATMENT AREA
PLANT 6 SUMP TREATMENT AREA
PLANT 5 SUMP TREATMENT AREA
PLANT 4 SUMP TREATMENT AREA
PLANT 1 SUMP TREATMENT AREA
DECONTAMINATION AREA
SOLIDS TO WASTE PIT VIA TRUCK
WATER TREATMENT PLANT EFFLUENT
STORM SEWER
TREATED SANITARY SEWER
HOLD TANKS
SLURRY TANKS
GENERAL SUMP
MANHOLE 175
RIVER
SAMPLER
PUMP HOUSE
CHEMICAL WASTE PIT
REFINERY SUMP TREATMENT AREA
LAB SUMP
PILOT PLANT SUMP
ORE REFINERY PLANTS 2 & 3
COLD SIDE
HOT SIDE
SUMP
NITRIC ACID RECOVERY AREA
RAFFINATE EVAPORATOR
NEUTRALIZING TANKS
COMBINED RAFFINATE AREA

PLATE NO. 3

PLATE NO. 4

PLATE NO. 3

SCALE NONE G-5013

to find satisfactory methods for storage or disposal. Most of the non-burnable, solid materials--such as glass, graphite, ceramic materials and concrete--have been disposed of in the Number 1 chemical pit. In storing these materials in the pit, segregation of wastes with respect to estimated potential recovery value has been maintained.

Burnable, contaminated wastes are burned in an incinerator at the plant site. A collector for cleaning off-gases has not been necessary. The ash from the incinerator has a sufficient uranium content for recycle to the recovery plant.

Satisfactory methods of disposing of contaminated organic solvent and contaminated oil have not as yet been found. These materials come from the extraction process and uranium machining operations, respectively. Investigation of possible disposal methods is continuing.

Control Limits for Disposal of Waste Water

The Fernald Plant has established standards for release of waste water based on recommendations of the Health Department of the State of Ohio and on National Bureau of Standards publications.(45) Permissible concentrations of uranium, radium, and alpha and beta activities in the plant effluent depend on the day-to-day river flows. Even distribution in the river of contaminants in the effluent over a 24-hour period is assumed. This assumption is justifiable because of the low order of contamination, our method of operation, the nature of the receiving water body, and the anticipated usage of the river water. Standards pertaining to the concentrations of radioactive components in the river are: uranium - .35 mg/l, radium - 4×10^{-8} μc/ml, Alpha activity - 1.0×10^{-7} μc/ml, and Beta activity - 1.0×10^{-7} μc/ml.

Water Survey Methods

The waste effluent to the river is continuously sampled by a wier-type sampler. The collected sample is removed and analyzed daily. The location of the sampler in the combined stream from the storm, sanitary and process water streams is shown in the plates. A second sampler is being installed at the outfall from the waste pits. This sampler will be combined with a meter to measure the volume of wastes discharged by this route.

A gaging station has been provided for measuring the river flow just below the plant outfall. Daily measurements are made of the flow to be used as a guide in waste operations. Daily records of the river flow, waste volume, analytical results and calculated concentrations for the various materials of interest are kept for reference.

Data Summary

Over the five-year period that waste disposal records have been kept, there has never been a case when the calculated uranium concentration in the river, averaged over a 24-hour day, exceeded .35 mg/l. The highest monthly average uranium concentration (calculated) was .039 mg/l. In the year 1958 the highest monthly average concentration was .005 mg/l.

Although the removal of alpha and beta activity from liquid wastes has not approached that of uranium removal, the ultimate goal of not more than 10^{-7} μc/ml for total alpha or total beta over any 24-hour period is a near achievement. During the last ten months of 1958, there were twelve days when the beta activity exceeded the target level. The highest calculated monthly average for the last ten months was $.54 \times 10^{-7}$ μc/ml for beta activity and $.04 \times 10^{-7}$ μc/ml for alpha activity. Further improvement of this record is anticipated in 1959 with use of the chemical pit for removal of suspended solids.

Considering the nature of the radioactive waste constituents, the control limit for beta activity is believed to be conservative. It is known that Thorium^{234} contributes most of the activity. The usual recommended maximum permissible concentration for this isotope is much higher than 10^{-7} μc/ml.

Cost Data

The actual cost of waste treatment at the Fernald Plant cannot be assessed. There has not been a detailed study of these operations for the purpose of providing accurate cost estimates. It is assumed that, disregarding contamination control requirements, the treatment of liquid wastes in the individual plants to recover uranium is economically justifiable. These costs are charged to production. No attempt is made in this report to separate them from other production costs. The same reasoning is applied to the cost of operating and maintaining the raffinate treatment equipment and storage tanks.

The following construction and operating costs are given for comparison with other methods of waste treatment and storage.

Capital costs include design, purchase and installation of listed equipment and construction costs of pits. No overhead or contingencies are included in operating costs. Because of the short time permitted to derive the operating costs, an inaccuracy of 25% more or less than the figures given may exist.

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FMPC Waste Material Handling and Storage Costs

Facility	Capital Cost	Operating Cost
General Sump		
Original installation (3-20,000 and 1-5,000 gallon tanks)	$100,170	
Expansion installation (3-50,000 gallon tanks)	$124,000	
Direct Labor and Misc. Maintenance		$4,000/month
Waste Pits		
Pit Number 1 (8.4 million gal. capacity)	$ 48,600	
Pit Number 2 (2.7 million gal. capacity)	$ 25,600	
Pit Number 3 (43 million gal. capacity)		
Pit and Spillway	$ 75,000	
Pump, tank, pipelines	$ 95,000	
Direct Labor to transport MgF_2 to Pit		$2,000/month
Concrete Storage Tanks		
For Radium Cake (2-1,000,000 gal. tanks)	$249,000	
For Metal Oxides (2-1,000,000 gal. tanks)	$210,000	

Relations with State

By mutual agreement with the Sanitary Engineering Division of the Ohio Department of Health, standards for waste control have been established for the Fernald Plant which equal or better other local, state, and national standards. The State-Company relationship of cooperative effort is recognized by the Company as an invaluable aid to achieving a high degree of success.

References:

1. Ohio Department of Health, Report of Water Pollution Study of Miami River 1949, issued 1951

2. Dove, George D. and Stanley E. Norris of U. S. Geological Survey Ground Water Branch, Columbus, Ohio, "Conditions Governing the Occurrence of Ground Water in the Fernald Area, Ohio, with Reference to the Possibilities of Contamination by Disposal of Chemical Wastes", September, 1951

3. Heatherton, R. C., "Disposal of Fluoride Wastes", TID-7517, Part Ia, October, 1956

4. National Bureau of Standards, Handbook 52

5. National Bureau of Standards, Handbook 61

STATEMENT OF TRACERLAB, INC., ON WASTE MANAGEMENT OPERATIONS

By SAMUEL S. AUCHINCLOSS, President, Tracerlab, Inc.

Tracerlab, Inc. since its founding in 1946 has become the leading supplier of the most diverse range of instruments, radiochemicals and services in the nuclear field.

From the company's very inception, a matter of great concern has been what to do with the radioactive leftovers. The procedures first used by Tracerlab and still in use today are rather costly and may soon become excessively burdensome because of the growing use of radioactive materials.

Tracerlab is in a unique position to observe the growth of the problem of waste disposal. Unlike any other single firm in the nuclear field, Tracerlab has the broadest view of waste disposal and decontamination problems owing to the fact that it is active in every single phase of applied radiation - hospital waste, laboratory waste, industrial waste, to name a few, all fall within our experience. Whether such waste be the end product of experimental use, the by-product of isotope production and purification, spent isotopes used in process control, we have been consulted on disposal or have physically done the disposal work ourselves.

In addition, Tracerlab holds the first radioactivity decontamination and disposal license issued by the AEC. Under its terms, we can decontaminate radioactive material which has been spilled and store, transport and arrange for the disposal of the resulting contaminated materials in any part of the United States. We have recently made a proposal to provide training for personnel, and mobile facilities for the decontaminatio

of ship reactor components units and to provide for the safe disposal of the resulting waste.

In short, we have watched the problem of waste disposal build up in ten short years from one which affected a handful of professional people dealing with tracer quantities of isotopes to one which affects thousands of people handling every conceivable quantity of isotopes; we have concluded that present methods leave much to be desired.

However, pending the discovery of better methods through the efforts of groups such as this committee, we feel that the problem must be faced with the same care as heretofore. In addition, an increasing number of people must be trained in decontamination and waste disposal to keep pace with the rapidly growing list of isotope applications and users. Failure to continue to provide competently trained personnel can only lead to disasterous consequences. We are certainly gratified to know that AEC issues licenses to waste disposal firms with the same degree of care as those for other types of operation.

Today, Tracerlab disposes of its radioactive solid waste, e.g. glass ware, paper towels, bench paper and equipment that has become contaminated in the following manner:

The solid waste is put into 35 gallon steel drums, and, for economic reasons, the drums are filled as completely as possible. These drums are put into ICC specified containers, powdered asbestos is placed between the two drums, and then

they are shipped via rail or truck to Oak Ridge for burial. Complete records are kept as to what the content of each barrel is, and all shipping is in accordance with I.C.C. regulations.

For disposal of radioactive liquids, this procedure is followed:

The radioactive liquid wastes are accumulated in one gallon bottles and these bottles are placed in a 35 gallon steel drum. The many bottles are cushioned by powdered asbestos which prevent movement and possible breakage. The spaces between the drums are further encased in concrete to insure sinking. Then a local sea disposal company which is licensed by the AEC is called in for pick up and disposal in a location approved by the AEC.

If it would be practicable, we at Tracerlab feel that a clear distinction should be made between the rules and regulations for disposal of radioactive waste generated by small commercial firms and users as compared to wastes generated from reactor type operations. The costs of waste disposal in this competitive field is a significant factor in the cost for the ultimate user.

The cost of packaging and disposing of the waste is high, and it does create a problem for the small user who has very little amounts of radioactive waste to get rid of. As the users of radioactivity increase, it is certain that

state laws will be incorporated defining their own requirements in matters pertaining to the disposal of radioactive waste, thus adding more to an already heavy burden.

To reduce this packaging cost, we think that the regulations should be based on the amount involved, the type of material, and on the half-life of the activity to facilitate disposal and reduce costs.

With the improvement of technology, the production of isotopes has increased and at the same time their prices have gone down. Strong efforts have been made in the last few years to expand the use of isotopes and to provide a larger market for them; but there is no corresponding attempt to make it easier for the potential user to dispose of waste, and especially of the tracer quantities of isotopes, in a more economical fashion. This striking imbalance between the pressure to use isotopes on the one hand and the restrictions on their disposal on the other hand is a major detriment to future expansion.

The nuclear industry in the United States has an outstanding safety record; one of the reasons for this record is that from the outset we have operated under stringent measures for producing, handling and disposing of radioactive materials. On the other hand, the safety records of England, and Canada are also good, but they have been achieved without the

excessively stringent disposal regulations of our country. The English, for example, have used the oceans far more freely to dispose of relatively high level wastes without requiring, in many instances, any form of container and without the requirement of a thousand fathom depth at the point of dumping. We certainly do not recommend promiscuous disposal of radioactive materials, but we do feel the time has come to reappraise the hazards in view of the greater knowledge of those hazards and the desire to employ radioactive material more widely. In contrast, at the present time our regulations allow the disposal of only minute amounts of radioactive wastes through uncontrolled carriers such as sewage, air, etc. in amounts not to exceed established limits.

For example, it might be feasible that the practice of off-shore dumping in specified dumping grounds of radioactive wastes of perhaps up to 1 curie per year per disposer in any one dumping be subject only to the restrictions that the container be of a type that will certainly sink and that the materials contained therein stay submerged until they are dissipated into the ocean.

Ocean going liners offer another plausible and less expensive way of disposing of containers housing radioactive wastes. Under the supervision of the AEC the liners could take the containers, properly prepared in accordance with I.C.C. regulations and drop them into the sea in the course of their

voyage.

It is gratifying to see the AEC take steps to assure a successful solution of the waste problem. Considering the vast scientific talents available to us today and the inevitable high future cost of continuing present safe disposal methods, we should take full advantage of this opportunity to re-examine the issue.

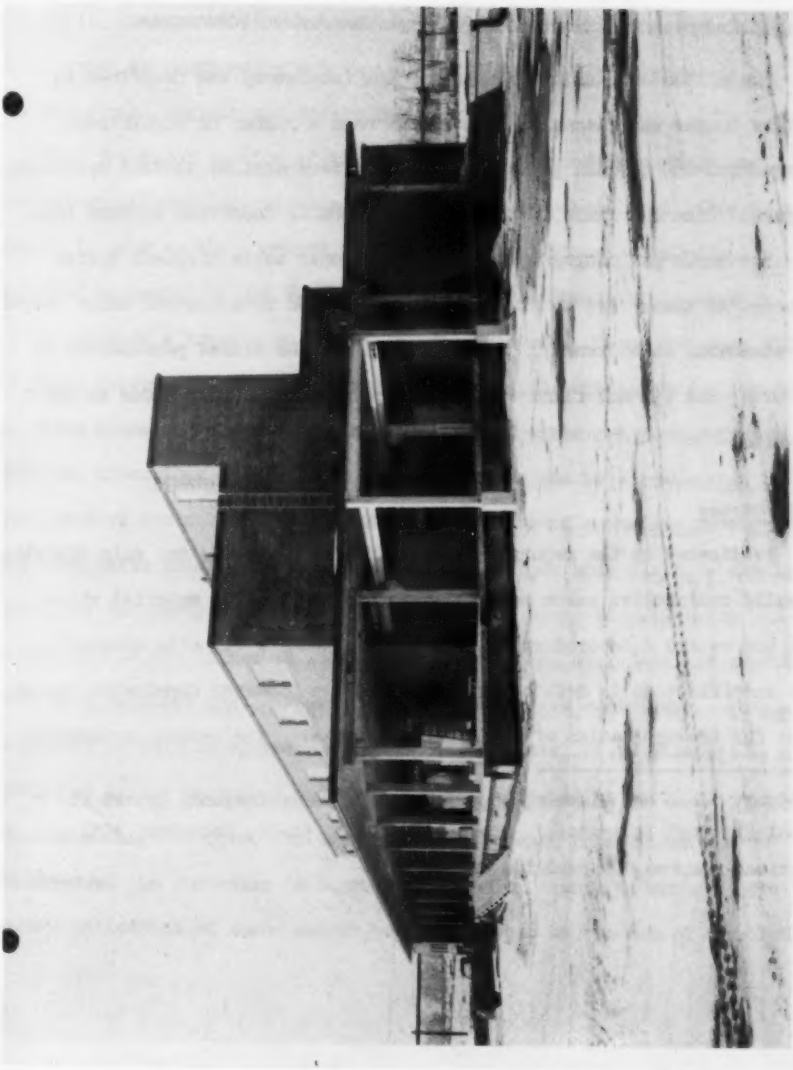
All present methods of waste disposal are quite primitive; the nuclear waste disposal problem faces us now the way stream pollution faced us thirty years ago, but we worked our way out of what seemed to be an impossible problem when we first started to work on stream pollution; so, we shouldn't give up too easily on the waste disposal problem. The possibility of a radical discovery in the area of waste disposal technology shouldn't be discounted in an age when we have discovered antimatter, when we have synthesized proteins, etc., etc. We should explore immediately even the most imaginative solutions. Of course, it may be that such work is already being carried out without the knowledge of people outside the AEC and the results may not be published because of security reasons, but assuming that the various government agencies are not already at work on the problem, we would urge them strongly to initiate a research program as quickly as possible.

MANAGEMENT OF RADIOACTIVE WASTES AT ARGONNE NATIONAL LABORATORY

Prepared by Members of the Staff of Argonne National Laboratory

Argonne National Laboratory, a research and development laboratory operated by the University of Chicago under a contract for the Atomic Energy Commission, is located on a 3700-acre plot 25 miles southwest of Chicago, Illinois. The Laboratory has about 3000 employees of which about 800 are staff members.

The Laboratory moved to its present site during the years 1950 through 1952. This location is in a highly populated area the ground water of which receives heavy use. In setting up the site the Laboratory was obliged to develop complete waste disposal facilities of all types. The anonymity of a large municipal sewer system was not available for dilution of the Laboratory outfall. That the technical and public relations aspects of the problems of waste disposal were well recognized by the Laboratory administration at that time is evidenced by the fact that the third building commissioned on the new site (June 1950) was one set aside for the development of processes for handling the radioactive wastes from the entire site. This building, called Bldg. 310 (see Figure 1), contains about 5000 sq ft of pilot plant development laboratory area (25 ft of



headroom), four laboratories, a counting room and office and service space. It cost about $1,500,000. The building was used during the period 1950 through 1953 for waste disposal development. Part of it, and part of the original development pilot plant equipment, is still being used for the routine processing of laboratory radioactive wastes. Approximately half of the building is now devoted to other uses.

The system originally planned for the laboratory was described in 1951 by Rodger and Fineman[1]. There have been a number of significant changes but the overall pattern now used is very similar to that originally proposed. The operation of the present system is described in this paper.

The basic philosophy upon which the Argonne waste disposal system is based is that: (1) no fluids will be released from control which exceed the standards established[2], (2) no wastes will be stored permanently on the site, and (3) all fluid wastes will be reduced to solids for shipment off-site.

Solid Wastes

Predicated by the requirement for off-site disposal, two main divisions of solid radioactive waste material exist. The first is material which will not exceed a laboratory imposed restriction for off-site shipment. This specification is defined by the Interstate Commerce Commission regulations for transportation of radioactive material. The second is material

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[1] Rodger, W. A. and Fineman, P., "A Complete Waste-Disposal System for a Radiochemical Laboratory," Nucleonics, 9, 6, 50-61, December, 1951

[2] National Bureau of Standards Handbook 52.

which does exceed the established restriction. At present, it is stored under controlled conditions to allow for natural radioactive decay. Although the responsibility for its handling is borne by the Waste Control Group of the Plant Services Division, the waste producer is required to supply complete information as to the origin and characteristics of the material. Assistance in the handling and disposition of the varied types of material is continually maintained by Radiation Safety Personnel of the Industrial Hygiene and Safety Division.

For the major portion of the solid waste produced throughout the Laboratory, a stainless steel container with a one-cubic foot fibre drum insert is used as the standard waste receptacle. These containers, supplied as needed by the Waste Control Group, are designed to operate by stepping on a treadle to operate a sliding cover. There are some 470 of these receptacles in approximately 200 locations within the Laboratory. In those areas where considerable quantities of waste are produced or the physical dimensions of the waste are larger than can be accommodated in the standard container, a 4 1/2-cubic foot cardboard container is supplied. For that waste which exceeds a 50 milliroentgen per hour limit, a shielded secondary and tertiary container is available. Since the shielding equivalent is two inches of lead, a dolly and a rollaway jack are provided with each shielded unit. Figure 2 illustrates these containers. When a container is full or approaches the radiation limit, it is surveyed by Radiation Safety personnel and the pertinent information recorded on a Hazardous Disposal form. In the event that special precautions are to be observed the container is tagged and the drum insert is marked. The normal collection of these wastes is accomplished by two men of the Waste

FIGURE 2.--Containers shown are used in the majority of situations for the collection of solid active waste. Illustrated is the primary fiber drum, secondary stainless steel, and assembly of shielded pots. The cap tube is utilized for high level waste disposal.

Control Group. Pickup from some 27 facilities in the Laboratory is maintained on a scheduled basis. It consists of removing the inner fibre drum insert, sealing the lid of the drum if required and removal from the area. Those wastes requiring shielding are handled essentially the same with the exception that they are transferred to additional shielded units. The day's collection is transported to the Waste Storage Area in a specially designed vehicle.

The Waste Storage Area (see Figure 3) comprises some 400,000 square feet of area in which the various control, concentrating and storage units are located. Admission to this area is restricted to operating personnel only. In this area all wastes are segregated into that which can be shipped offsite and that which requires storage. Material that does not exceed 200 mr/hr and/or does not contain any alpha active material other than natural uranium is baled. Concentration of about half of the solid waste is accomplished by compression in a commercial type hydraulic baling press (see Figure 4). Operating pressures of up to 2100 psi enable an average volume reduction factor of four. Coupled with the baler is a 400 cfm exhaust system which includes a two-bank filter unit. The baled material and other non-balable material is packaged into a final container for offsite disposal. This container, fabricated of 12-gauge steel, reinforced with angle iron, is 4 feet wide by 5 feet long by 6 feet deep. A steel lid with a built-in gasket when bolted to the bin results in a shipping unit of sufficient integrity to meet the requirements for interstate transportation on public carriers. Larger components of shippable waste such as 55-gallon drums of solidified liquid waste and major components of pilot operations, transported directly to the Waste Storage Area, are handled similarly to baled material.

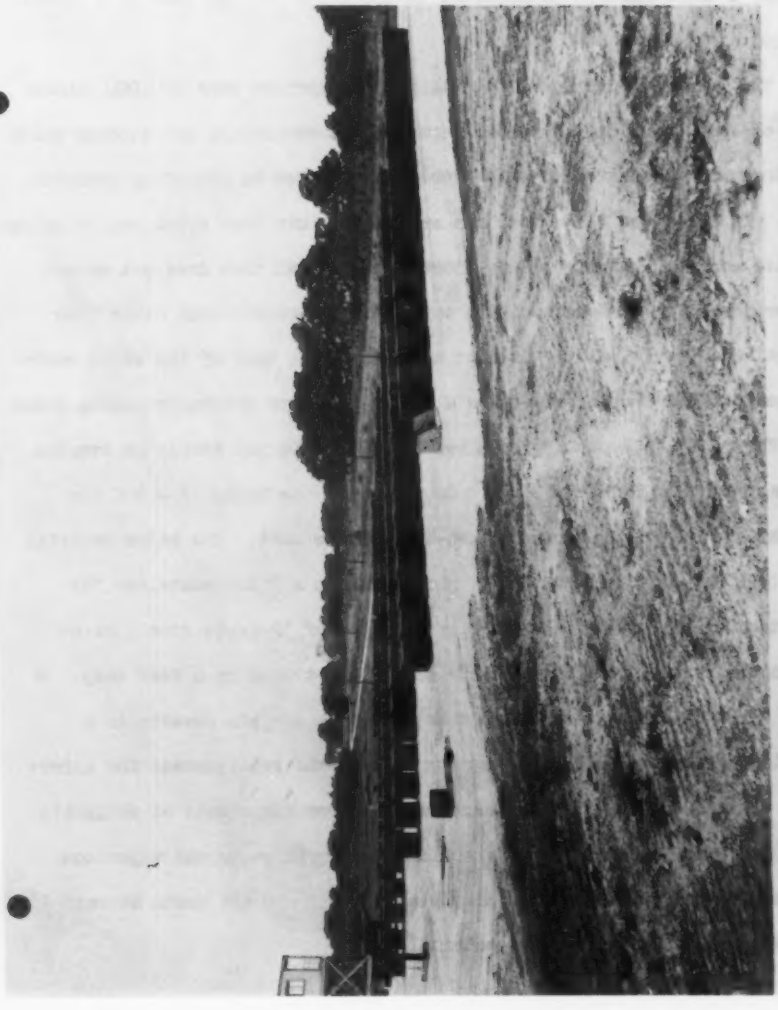

FIGURE 3.—North view of waste storage area 317. Area reserved for above ground storage of waste-filled shipping containers. In the background upper left is the boiler building and storage shed.

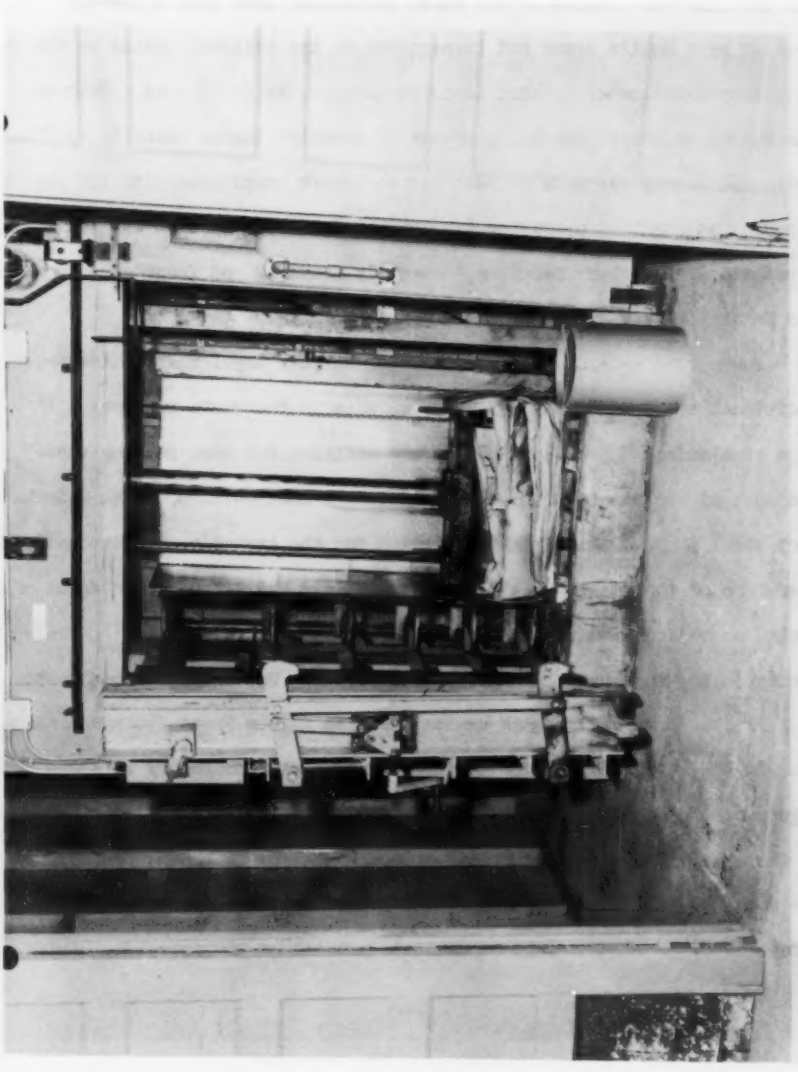

FIGURE 1.—Rear view of boiler and enclosure. Concealed hole illustrates the reduction of 16 cubic feet.

At present, one shipment of packaged waste is anticipated per year. In April 1957 a 240-bin, 12-car shipment was made to the Oak Ridge National Laboratory. Prior to shipment each bin was numbered, the average weight determined and a radiation survey made of the five exposed bin surfaces. The 240 bins were loaded at the Waste Collection Area onto a low-boy trailer by a mobile crane and transported to the railroad siding on the site. Here they were loaded by another crane into the railroad cars. Each car was shored to insure minimum movement of material during transit. A final radiation survey was made of each car to insure compliance with ICC regulations. The combined operation was completed in about 4 days utilizing personnel of the Plant Services, Materials Handling and Industrial Hygiene and Safety Divisions.

Due to the shielding required for the higher level radioactive waste, additional types of containers are available to the waste producer. A type of aluminum map case (Figure 2) is utilized for some reactor waste and most of the remote control cave waste. A horizontal pot with 8-inch equivalent of lead shielding is utilized for the transfer and disposal operation of this waste (see Figure 5). Thirty and fifty gallon drums of waste can be accommodated in a specially designed 10-ton coffin. Transportation to the storage area is provided by the Waste Collection Group.

Five underground storage vaults, located in the Waste Storage Area, provide the temporary storage requirements for the Laboratory as shown in Figure 6. Three of the vaults, constructed of reinforced concrete are in the shape of an open box with a drainage control on the bottom. These are about 95 feet long by 13 feet wide by 11 feet deep. They extend some 8 feet below grade, and will accommodate a total of 38 bins each. After

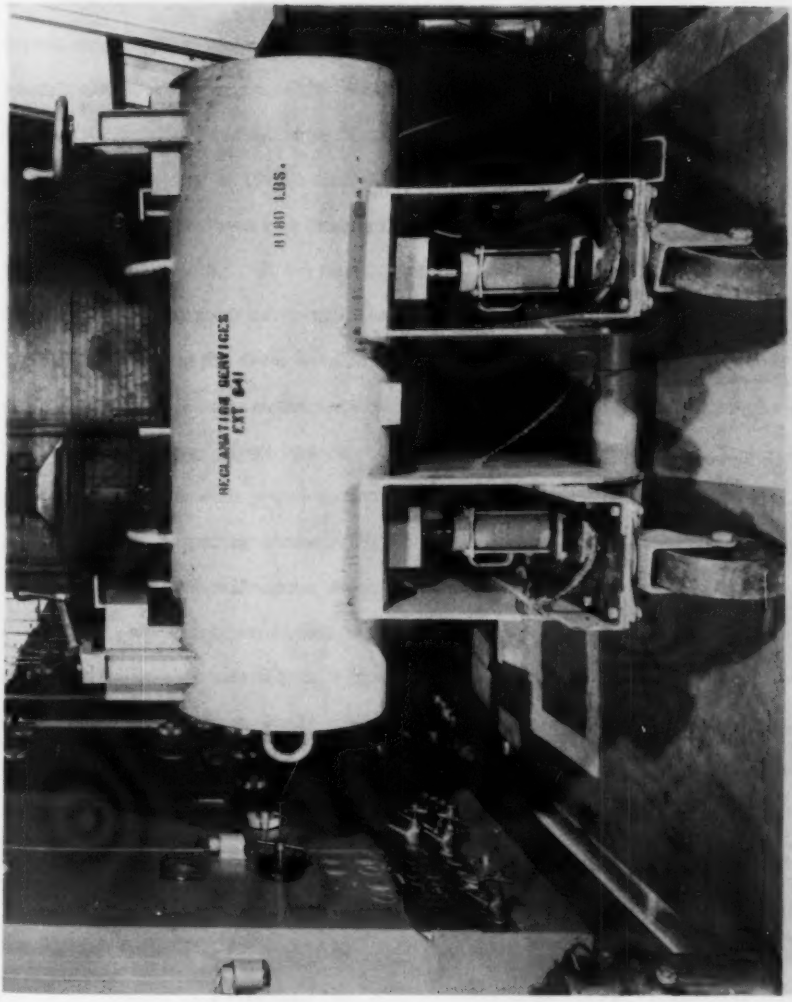

FIGURE 5.--Four-ton shielded pot utilized for the disposal of high level radioactive waste.

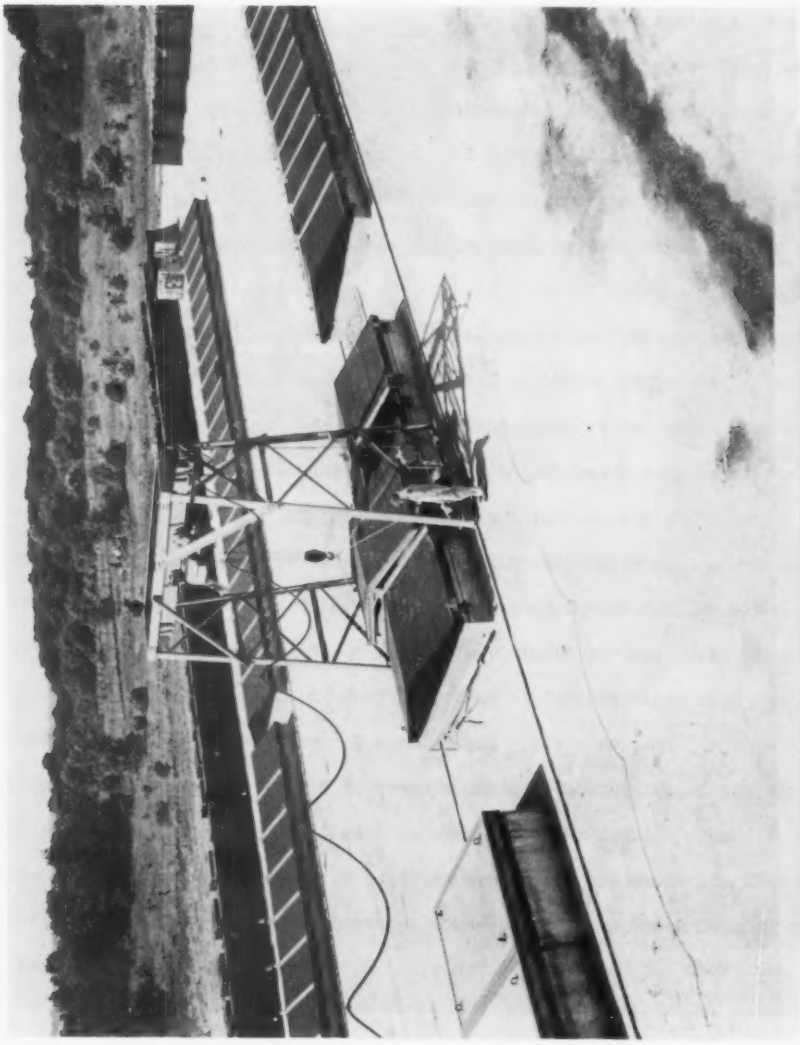

FIGURE 6.—South view of waste storage area 317. Includes on left, the 6-cell storage vault, middle, the tube storage vault, and the cask storage vault.

the bins are placed in the vault, a 1-foot thick concrete lid is placed above the bins. Each lid has two holes which allows the disposal of waste into the respective bin. After a bin has been filled, the hole is plugged. Protection from the weather is provided by peaked wooden roof sections covered with tar paper. Approximately five years' storage space exists in these vaults, assuming the current rate of waste production. The fourth vault, of reinforced concrete, constructed in the open box shape, consists of six adjacent cells, each cell 6 feet by 10 feet in area by 21 feet deep. The vault, some 18 feet below grade, is utilized for storage of bulky and highly active items of waste. Each cell has its respective cover which must be removed prior to the vault's use. The fifth storage vault is an assemblage of vertical pipe casings, 4-inch to 10-inch diameter, cast as part of a concrete slab. Its dimensions are 12 feet by 30 feet in area by 21 feet deep with 19 feet below grade. The pipe casings serve as receptacles for the aluminum map case and similar sized containers. Six-inch thick lead plugs are used to cap the exposed pipe casings. Some 9 to 10 years' storage space is available in the last two described vaults. A 5-ton capacity gantry crane provides the means for the handling of vault lids, roofs and miscellaneous equipment required for storage operations. It directly services the 6-cell vault, tube vault and one-bin vault. The crane controls are located in a tower, 26 feet in height and about 60 feet from the 6-cell storage vault. This allows for the remote control handling of the crane in connection with the storage of high level waste. Various types of remote control equipment such as drum tongs, various size chutes, guide rods and accessory slings are utilized in these operations.

A tabulation of total solid waste representing the accumulation for the Fiscal Year 1957 is shown in Table 1. It is separated into the various

operations and level of activities that are encountered in the waste disposal program at Argonne National Laboratory. Approximately 97 per cent of all solid wastes were considered low level and prepared for offsite disposal.

Table 1

ACCUMULATION OF SOLID WASTE
FY 1957

Total Volume Low Level Waste Collected	23,900 cu ft
Volume Low Level Waste Concentrated	10,200 cu ft [a]
Total Low Level Waste Packaged for Shipment	16,100 cu ft
Collection and Storage High Level	850 cu ft

[a] Compressed volume ~ 2400 cu ft

A block flowsheet of the solid waste disposal operations is shown in Figure 7.

During the years 1951 and 1952 combustible radioactive wastes were incinerated. An experimental incinerator of 100-cu ft/day capacity was designed and built with the assistance of the A. D. Little Co.

The incinerator equipment consisted of a type 330 stainless steel incinerator body in which the material was burned in the presence of excess air. Off gases from the incinerator were put through three liquid scrubbers to remove the larger particles of fly ash. Before discharge to the atmosphere the off gas was sent through an AEC filter. The ash resulting from combustion fell through a grate system and was collected in water

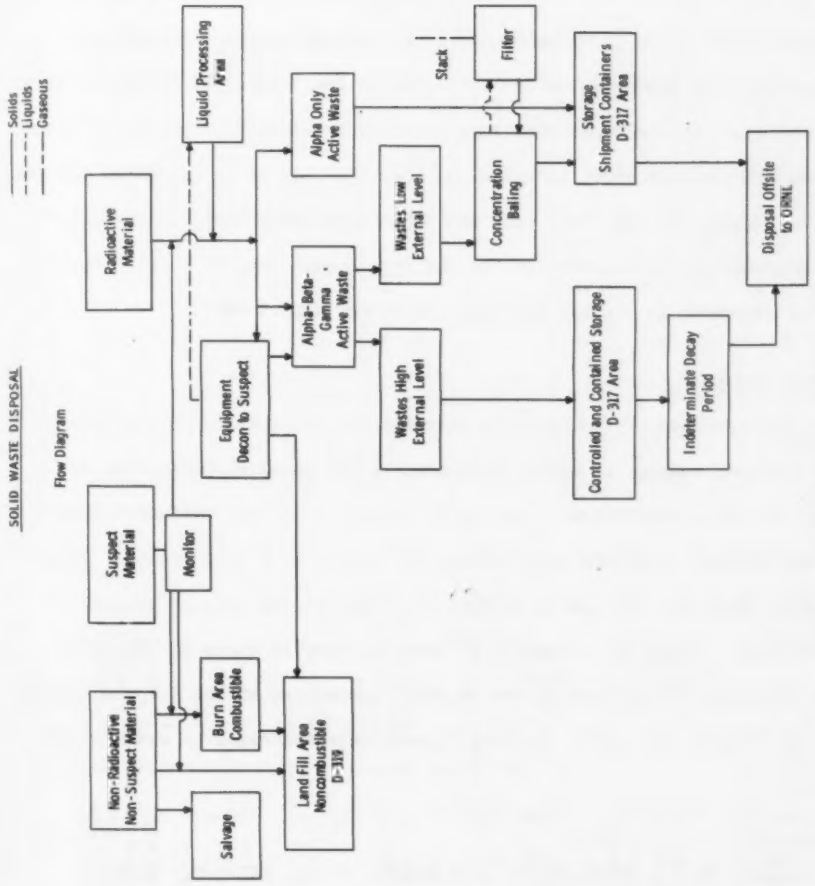

FIGURE 7.--Solid waste disposal flow diagram.

in a cone located in the base of the furnace proper. These ashes were finally collected in a canvas bag filter inside a stainless steel drum. The bag and its contents were removed from the system, devatered and then stored.

The incinerator operated well. It produced an ash volume of 5 per cent of feed. It was demonstrated that as little as 1 part in ten million of the radioactivity fed to the incinerator was released to the atmosphere. However, even at an installation as large as Argonne the volume of combustible waste produced was not sufficiently great to allow 24-hour a day operation of the unit and when long term arrangements for removal of solid waste offsite were made, it was clear that the cost of incineration did not quite justify its use. The unit has since been dismantled. A complete description of the incinerator and the development program carried out on it is contained in Argonne National Laboratory Report 5067.[3]

Liquid Wastes

The problems of liquid waste disposal and the systems to handle them are the most complex of all as evidenced by the greater requirement of manpower and expenditures. The varied properties of the different liquid wastes present a problem which cannot be handled in a unit system. Consequently there are four major systems in effect at the Argonne National Laboratory. A general schematic of these systems is shown in Figure 8. The following resume' covers the physical operations of the disposal system for the Fiscal Year 1956. Current figures have increased by about 20 per cent.

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3 Hampson, D. C., Hykan, E. H., and Rodger, W. A., ANL-5067, Basic Operational Report of the Argonne Active Waste Incinerator, February, 1953.

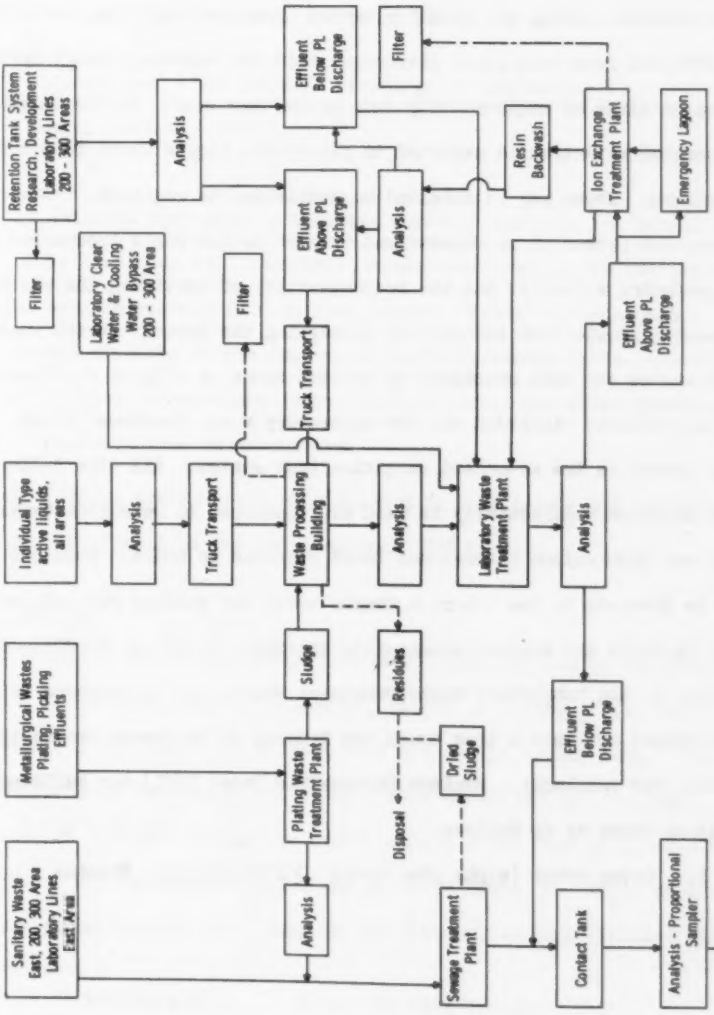


FIGURE 8.--Liquid disposal system flow diagram.

The first is the sanitary system which comes from both areas (research area or West Area and administrative area or East Area) and flows to the Sewage Treatment Plant the outfall of which goes to the Des Plaines River. No radioactivity is knowingly permitted in this system. The second system, a small one located in the East Area, is a plating waste treatment plant which handles plating and pickling wastes from the Metallurgy Division. The effluent from this plant discharges into the sanitary sewage system. In the handling of radioactivity two systems are used. In the first place, the working scientist is expected to put active liquid waste into supplied containers. These may be shielded or unshielded as required. These containers are picked up on request and trucked to the Waste Processing Building. The producing scientist has the responsibility of providing the waste processing people with information concerning the content of the waste. These wastes are then processed by various means as will be explained later and the effluent discarded via the Laboratory Waste Treatment Plant. The final system is the so-called retention tank system. All sink drains in areas in which radioactivity is used are conducted to retention tanks. These are 1500-gallon glass-lined tanks operated in pairs. When one is full, flow is diverted to the other, a sample taken and checked for radioactivity. If it is below the maximum permissible discharge level, it is discarded directly to the Laboratory Waste Treatment Plant. If it is above this level, it is pumped out into a tank truck and trucked to the Waste Processing Building for treatment. Maximum Permissible Level (MPL) for building retention tanks is as follows:

1. Gross count (alpha plus beta) $\geq$ 1000 dpm/ml: Process

2. Gross count < 1000 dpm/ml but ≥ 100 dpm/ml: Analyze for Sr^{90} and count for total α.
   a. Sr^{90} level ≥ 10 dpm/ml and/or α level ≥ 5 dpm/ml: Process.
   b. Sr^{90} level < 10 dpm/ml and α level < 5 dpm/ml: Discharge to Waste Treatment Plant.

3. Gross count < 100 dpm/ml: Discharge to Laboratory Waste Treatment Plant.

It is expected that most of the time activity found in the retention tanks will be less than MPL. Operating experience, as will be indicated later, has been reasonably good in this regard. Retention tanks below MPL are pumped directly to the Laboratory Waste Treatment Plant--essentially, four 70,000-gallon holdup tanks--where pH is adjusted and final discharge control on radioactivity is maintained.

Each of these systems will be discussed in more detail and operating figures for the fiscal or the calendar year 1956 given.

Sanitary System

The sanitary system and Sewage Treatment Plant consist of the following equipment: comminuter, Dorr Clarigester, trickling filter, final clarifier, sand sewage filters, chlorine contact tank and sludge drying beds. The plant was designed for an average 24-hour flow of 217,000 gallons/day, a peak rate of 570,000 gallons/day and a maximum storm rate of 1,300,000 gallons/day. The operating results are shown in Table 2. It is of interest to compare these results with those of the Chicago Sanitary District which are as follows:

BOD Reduction	90 to 95% (92% Average)
Effluent BOD	5 to 15 ppm.

Table 2

SANITARY SEWAGE TREATMENT PLANT OPERATION

July 1, 1955 through June 30, 1956

	Average Flow GPD	Average BOD in ppm		Average Reduction %
		INF	EFF	
July, 1955	374,000	58.9	1.8	96.9
August	344,700	72.2	1.5	97.9
September	295,000	50.8	2.1	95.9
October	311,700	34.7	1.9	94.5
November	280,900	51.9	2.4	95.4
December	270,500	56.2	3.1	94.5
January, 1956	286,800	66.9	1.8	97.3
February	327,100	65.3	2.8	95.7
March	347,300	78.6	1.7	97.8
April	388,000	64.4	2.9	95.5
May	443,600	75.5	16.8[a]	77.7[a]
June	396,300	61.5	4.5	92.7
Fiscal 1956 average	338,825	61.6	3.6	94.1

[a] Sand filters by-passed during May for maintenance.

The design flowrate has been continuously exceeded. Nevertheless, the quality of the effluent has continued very high due chiefly to final intermittent sand filtration. The average BOD reduction of 94.1 per cent and the average effluent BOD of 3.6 ppm are well within operating goals. The actual discharge to the Des Plaines River is a combination of Sewage Treatment Plant effluent and Laboratory Waste Treatment Plant effluent

discharged at an average ratio of 2.1 to 1.

The effluent BOD from Laboratory Waste Treatment Plant varies between 0.6 and 45 ppm and averages 10.2 ppm. It follows that the BOD of the combined discharge averages 5.7 ppm. This discharge is within operating goals and meets the state criteria for sewer effluent discharges to the Des Plaines River.

Composite samples of the sewage plant effluent of the digested sludge are regularly checked for activity. No activity above MPL has been detected in the sewage plant effluent to July 1, 1957. Activity above MPL has been found in the digested sludge, requiring its disposal as solid active waste.

Plating Waste Treatment Plant

A small plating waste treatment plant is located in the East Area to handle some special metallurgical wastes. The plant contains three batch retention tanks which are used for plating wastes, cyanide wastes and pickling wastes. The principal equipment consists of recirculation pumps, agitators, a chlorine feeder, a sulfur dioxide feeder, a caustic feeder, filters and pH indicating and control equipment.

Removal of cyanide is accomplished by chlorination with basic pH control. Chromium is removed by treating with sulfur dioxide and caustic; acid wastes are merely neutralized. Copper and heavy metals are removed by precipitation with sodium hydroxide. The effluent from this plant is discharged to the Sewage Treatment Plant. The chief goals of this plant are to prevent the discharge of copper, chromium and heavy metals in quantities that would affect the biological operation of the sewage plant, to reduce cyanide to nontoxic constituents, and to adjust the pH.

The results of the operation of the plant for the fiscal year 1956 are shown in Table 3.

Table 3

ARGONNE NATIONAL LABORATORY
INDUSTRIAL WASTE TREATMENT PLANT

July 1, 1955 through June 30, 1956

Period	Gallons of Waste Solution	Average Activity (dpm/ml)		Chromium (ppm)	
		INF	EFF	INF	EFF
July 1 -24	15,843	5.4	3.9	--	--
July 25 - ..ugust 21	24,668	14.1	10.8	1.1	1.1
August 22 - September 25	27,131	3.7	4.2	1.1	0.3
September 26 - October 23	26,920	24.1	22.2	0.4	0.4
October 24 - November 20	19,330	12.4	12.2	0.1	0.1
November 21 - December 25	26,239	7.5	6.6	6.8	0.7
December 26 - January 22	18,944	16.4	27.3	0.0	0.0
January 23 - February 19	29,354	11.2	10.4	3.1	3.1
February 20 - March 25	37,200	22.5	9.7	650	5.0
March 26 - April 22	98,215	65.6	8.1	210	3.2
April 23 - May 20	25,169	12.6	6.1	3.6	3.6
May 21 - June 30	67,575	5.1	2.5	37.5	5.0

Total volume for period 416,587 gallons

No copper detected during the above period.

No cyanide treatment made during the above period.

Treatment of Active Liquid Wastes

The operating philosophy of the active waste treatment system at Argonne is based upon a high degree of segregation at the point of production of the waste. The operating scientist is asked to put any wastes into containers which are supplied to him upon request and removed whenever he desires. He is requested to provide complete information concerning the chemical and radiochemical constituents of the wastes that he has produced. These are then trucked to the Waste Processing Building. The total volume thus collected each month seldom exceeds a thousand gallons.

No radioactivity is knowingly permitted in the Laboratory drains. However, to provide complete control of Laboratory effluents, the drains in active working areas are connected to retention tanks. These are glass-lined tanks of 1500-gallon capacity. They operate in pairs (see Figure 9). When one is full, flow is diverted to the other, a sample taken and checked against the MPL shown earlier. If the radioactivity found is below the MPL, the waste is pumped directly to the Laboratory Waste Treatment Plant. If above MPL, the water is pumped into a portable tank and trucked to the waste processing area. The trucking equipment is shown in Figure 10.

During the fiscal year 1956, 98 per cent of the retention tanks were below MPL and discardable directly. The operating experience is indicated in Table 4.

Active waste collected in various pots and containers and the above-tolerance retention tanks are treated at a central processing facility (Building 310) wherein there is equipment for evaporation, filtration, flocculation, concrete solidification, vermiculite absorption, solvent washing and ion exchange. The collected wastes are treated in whatever manner will give the needed decontamination at the least cost. Operating experience for the calendar year 1956 is shown in Table 5.

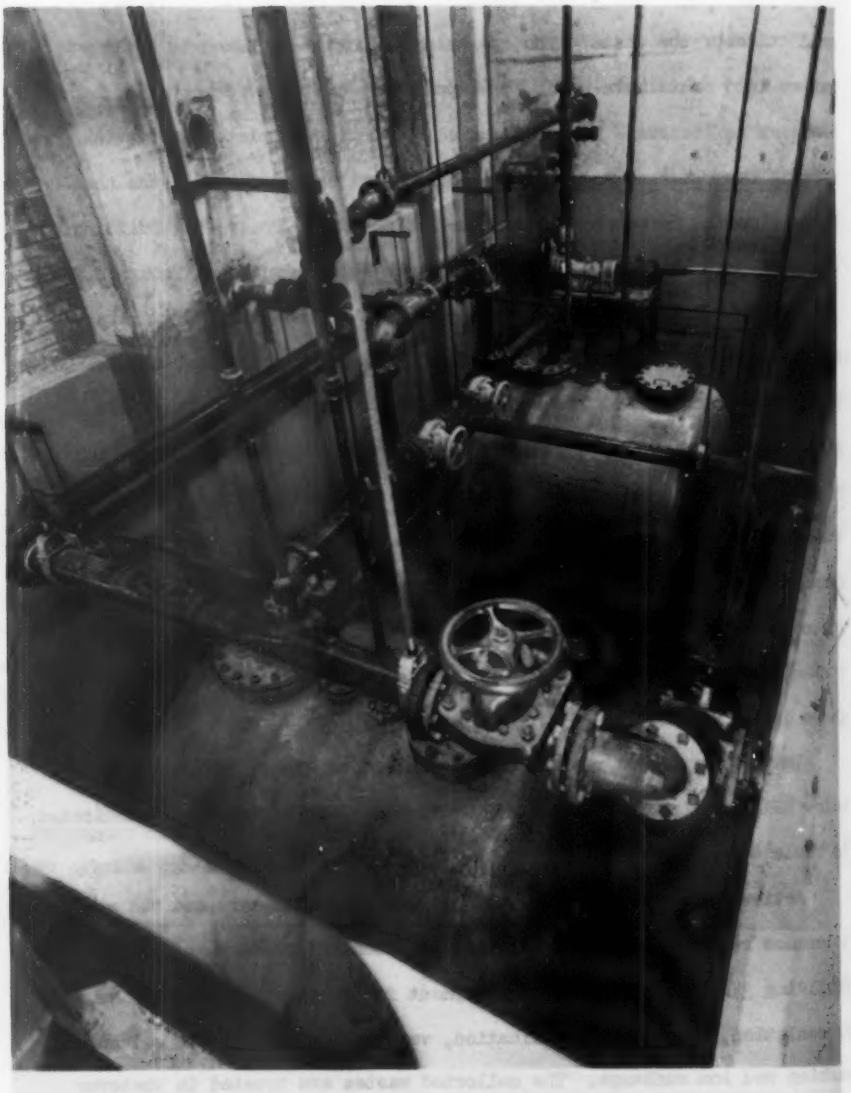


FIGURE 9.--Typical building retention tank installation. The connections between tank units allow for the continuous use of laboratory drains.

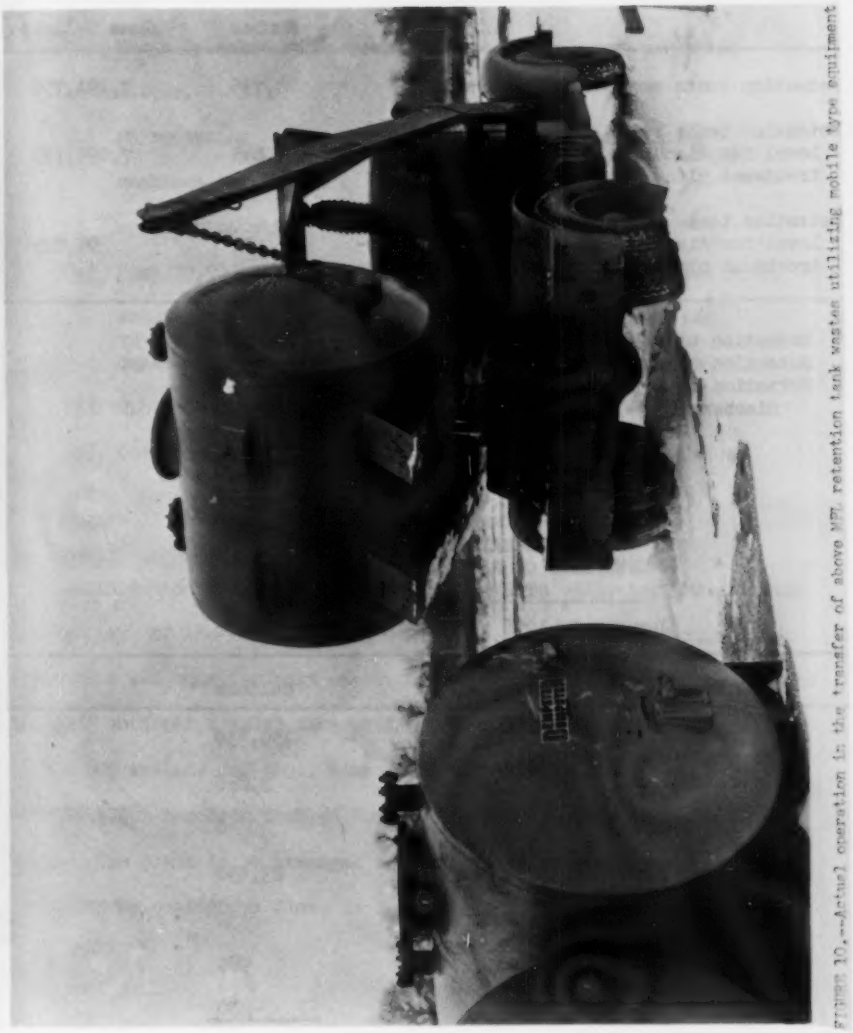


FIGURE 10.--Actual operation in the transfer of above MW retention tank wastes utilizing mobile type equipment

Table 4

ARGONNE NATIONAL LABORATORY
OPERATION OF LABORATORY DRAIN WASTE SYSTEM

July 1, 1955 through June 30, 1956

	Number	Volume Gallons
Retention tanks sampled and analyzed	4,736	7,191,750
Retention tanks below maximum permissible level for discharge to laboratory waste treatment plant	4,645	7,097,725
Retention tanks above maximum permissible level for discharge to laboratory waste treatment plant	91[a]	94,025

[a] Retention tanks above gross permissible discharge level 59
Retention tanks above alpha permissible discharge level 24
Retention tanks above strontium-90 permissible discharge level 8

Table 5

ARGONNE NATIONAL LABORATORY
OPERATION OF RADIOACTIVE LIQUID WASTE TREATMENT BUILDING

January 1, 1956 through December 31, 1956

Waste Treated by:	Gallons
Evaporation	159,330
Filtration	3,850
Ion Exchange	0
Flocculation	23,450
Concrete Solidification	30
Vermiculite Absorption	980
Solvent Washing	280
Total	187,920

These treated wastes are also discharged to the Laboratory Waste Treatment Plant along with the retention tank wastes which were determined to be below MPL and with cooling water which by-passes the retention tanks.

The Laboratory Waste Treatment Plant consists of the following principal units:

(a) Manually raked bar screen for screening coarse suspended matter.

(b) pH recording and control equipment, dry chemical feed machines, mixing tank; for automatic pH correction by means of caustic addition.

(c) Four 70,000-gallon tanks containing recirculation facilities and sludge scraper mechanism for equalizing the pH and collecting the settled sludge.

(d) One rate controller to control the discharge flow.

(e) One 385,000-gallon capacity earth lagoon for emergencies.

Figure 11 shows a portion of this system.

Disinfection is accomplished by chlorination utilizing the chlorine contact tank of the Sewage Treatment Plant. The plant was designed to neutralize and equalize acid wastes by means of caustic addition. It was designed to treat a total flow of 137,000 gallons per day at a peak rate of 420,000 gallons per day. Since this time the original tank capacity of this plant has been doubled from two to four 70,000-gallon holding tanks. The plant is so arranged that it can be operated by the batch method or by continuous flow.

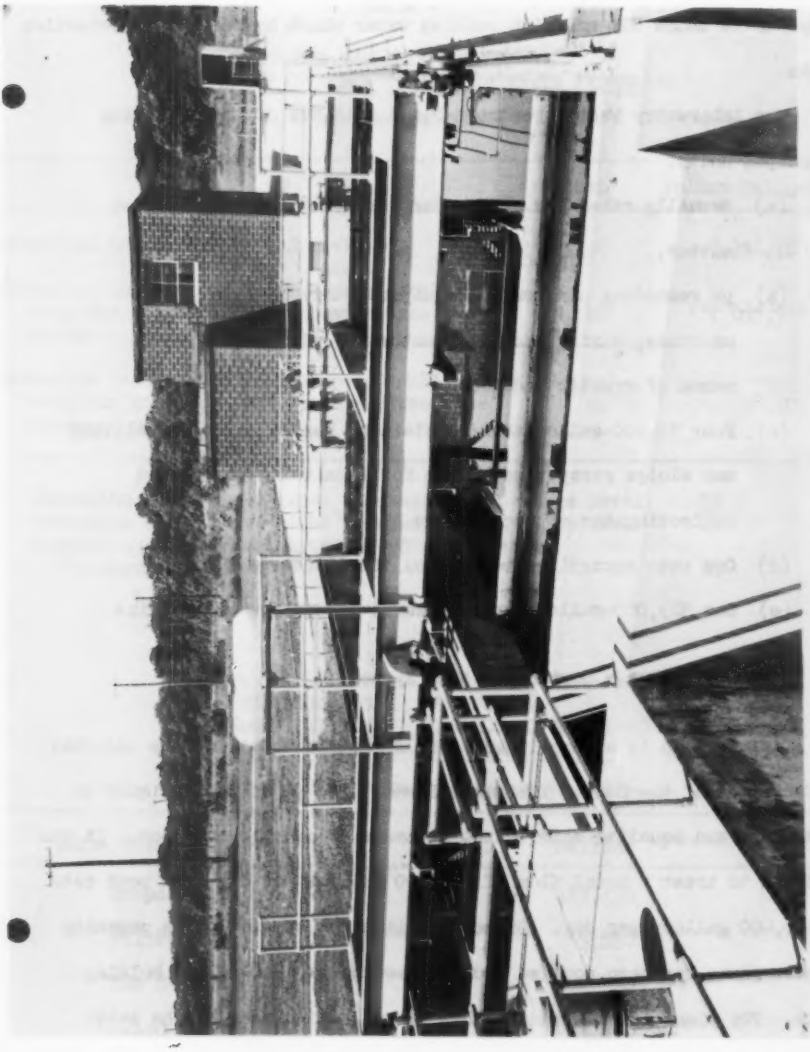


FIGURE 11.—South half of the laboratory waste treatment plant.

The MPL for final effluent discharge has been established at the following rates per year:

Type of Activity	Yearly MPL
Alpha Emitters	0.15 curie
Beta Emitters (other than Strontium-90)	10 curies
Strontium-90	0.45 curie

Provisional operating MPL for the Laboratory Waste Treatment Plant are as follows:

1. Gross count (alpha plus beta) $\geq$125 dpm/ml:
   Hold for process.
2. Gross count $<$125 dpm/ml, but $\geq$5 dpm/ml and/or α count $\geq$ 1 dpm/ml: Analyze.
   a. Sr^{90} level $\geq$ 2 dpm/ml and/or α level $\geq$1 dpm/ml: Obtain ruling from Industrial Hygiene & Safety Division as to disposition.
   b. Sr^{90} level $<$2 dpm/ml and α level $<$1 dpm/ml: Discharge to outfall.
3. Gross count $<$5 dpm/ml and α count $<$1 dpm/ml: Discharge to outfall.

Using a total volume of effluent discharge per year at the outfall of 150×10^6 gallons as a dilution factor, the yearly concentrations with the above limits could be:

1. 1.33×10^{-7} μc/ml of α emitter.
2. 4×10^{-7} μc/ml of Sr^{90}.
3. 1.67×10^{-5} μc/ml of β emitters, other than Sr^{90}.

It should be noted from U. S. Department of Commerce Handbook 52, that the only maximum permissible concentrations which could be exceeded using these MPL values would be those for Ra^{222} + 1/2 dr and At^{211}. However, it is felt that these isotopes make up only a very small fraction of the total radioactive contaminants discharged. The additional dilution contributed by the actual discharge (198 x 10^6 gallons for fiscal 1956) and by the Des Plaines River, which has not been included in the calculations, further reduces the concentrations of radioisotopes.

Operating results for the fiscal year of 1956 are shown in Table 6.

Gaseous Wastes

The bulk of the gaseous wastes comes from hood and cave exhausts. Chemical, metallurgical or biological operations involving radioactive materials, as well as operations using hazardous although non-radioactive materials, are done in hoods. The exhaust from such a hood may run from one thousand to several thousand cubic feet per minute. In the Chemistry Research laboratory building alone, there are 175 of these hoods.

In the newer laboratories, the air which goes into the hood is conditioned and filtered, (see Figure 12). Easily accessible at the back of the hood are four prefilters that even out the air flow throughout the hood volume and serve as prefilters for both dust and radioactivity, thus lessening the load on the rest of the system. The exhaust gases are conducted through a final filter before they reach the blower that discharges them through 3-foot stacks on the roof.

Periodically, personnel of the Building Services Department check the systems to assure that the requirements for efficient filtering of discharged gases are met. Radiation Safety personnel periodically survey filters to insure that personnel working in close proximity are not exposed to

Table 6

ARGONNE NATIONAL LABORATORY
LABORATORY WASTE TREATMENT PLANT OPERATION
July 1, 1955 through June 30, 1956

Total number of Tanks Analyzed 1,111

Total Gallons Discharged 63,986,360

Number of Tanks Containing Activity Above Provisional MPL 2

Total Gallons in Above Tanks 123,770

Disposition: Both tanks were discharged to the Emergency Lagoon. The Emergency Lagoon contents were eventually discharged through the outfall sewer, the effluent of which was below provisional MPL as a result of (1) decay, (2) dilution due to rainfall and laboratory waste overflows into the lagoon.

Radioactivity Discharged to the Laboratory Outfall

Type of Activity	Yearly MPL	Activity Discharged July 1, 1955 through June 30, 1956
Alpha Emitters	150 millicuries	13.7 millicuries
Beta Emitters (other than Strontium-90)	10 curies	0.135 curie
Strontium-90	450 millicuries	0.42 millicurie

Minimum "pH" of Raw Waste	0.7
Maximum "pH" of Raw Waste	9.7
Minimum "pH" of Treated Waste	5.2
Maximum "pH" of Treated Waste	9.7
Minimum BOD Treated Waste[a]	0.6 ppm
Maximum BOD Treated Waste[a]	45 ppm
Average BOD Treated Waste[a]	10.2 ppm

[a] Results obtained from occasional checks of tank contents.

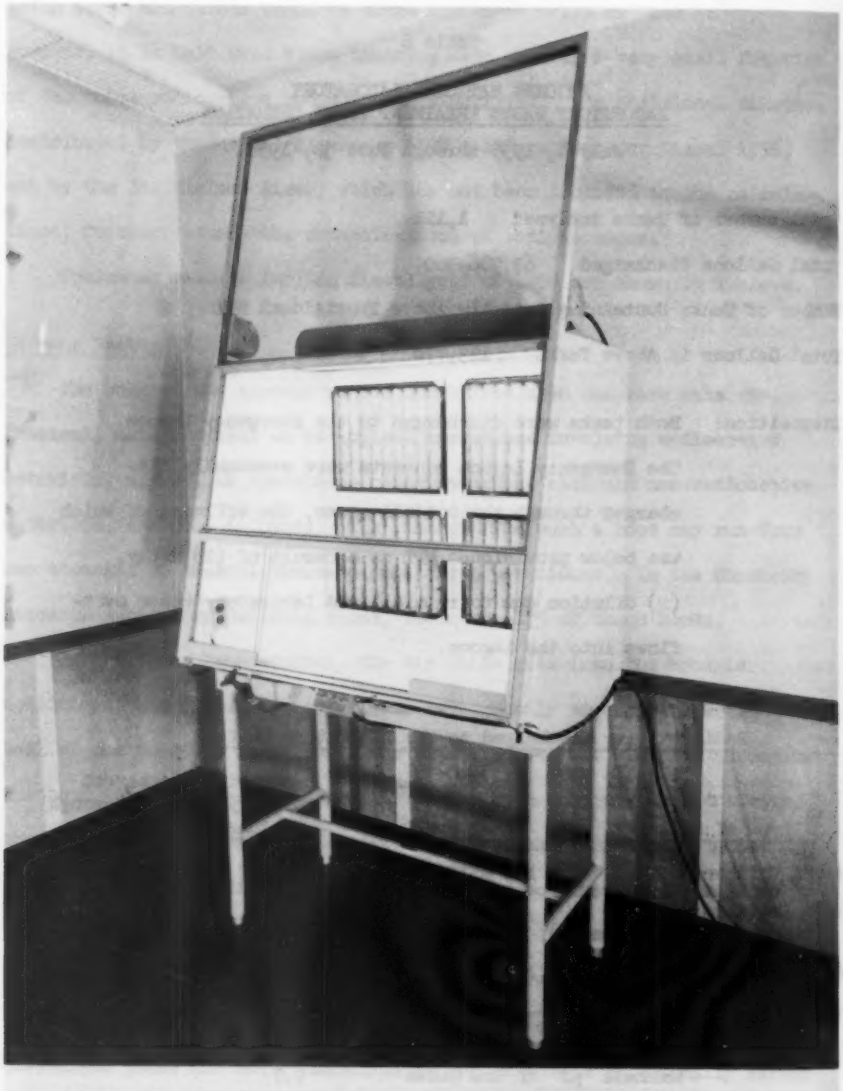


FIGURE 12.--Example of the ANL standard hood and table equipped with the typical restricted openings for access to do work in the hood.

radiation above the permissible level. Personnel of the Reclamation Department are requested to install new filters when particulate matter retained on the filter restricts the passage of air below the limit prescribed by design requirements or when the level of radioactivity rises above the permissible level. The removal and installation of filters includes the delivery of the spent filters to the Waste Storage Area where they are disposed of as solid waste.

Various filter media are available for use in prefiltering gases under a variety of conditions. In the majority of cases a fibre glass type filter is used. The final filters, now referred to as the AEC filter, are located in service areas adjacent to the exhaust stacks. For a typical exhaust system, see Figure 13. These filters are fabricated from CC-6 paper which was originally developed by the U S Army Chemical Corps for use in gas masks. It consists of fine asbestos fibers mixed with coarser cellulose fibers to give mechanical strength and serve as support for the asbestos. A noncombustible glass media final filter is used in some installations.

Tools used to machine uranium are each individually hooded. Metal and oxide dusts make up the bulk of the contaminants in the 1,200 cu ft/min discharged from each of these machines. These dusts are removed in a Rotoclone system. The Rotoclone consists of two sections; the first, a water impingement tank, the second, an electrostatic precipitator. The liquid scrubber waste is eventually trucked to the Waste Processing Building for further processing.

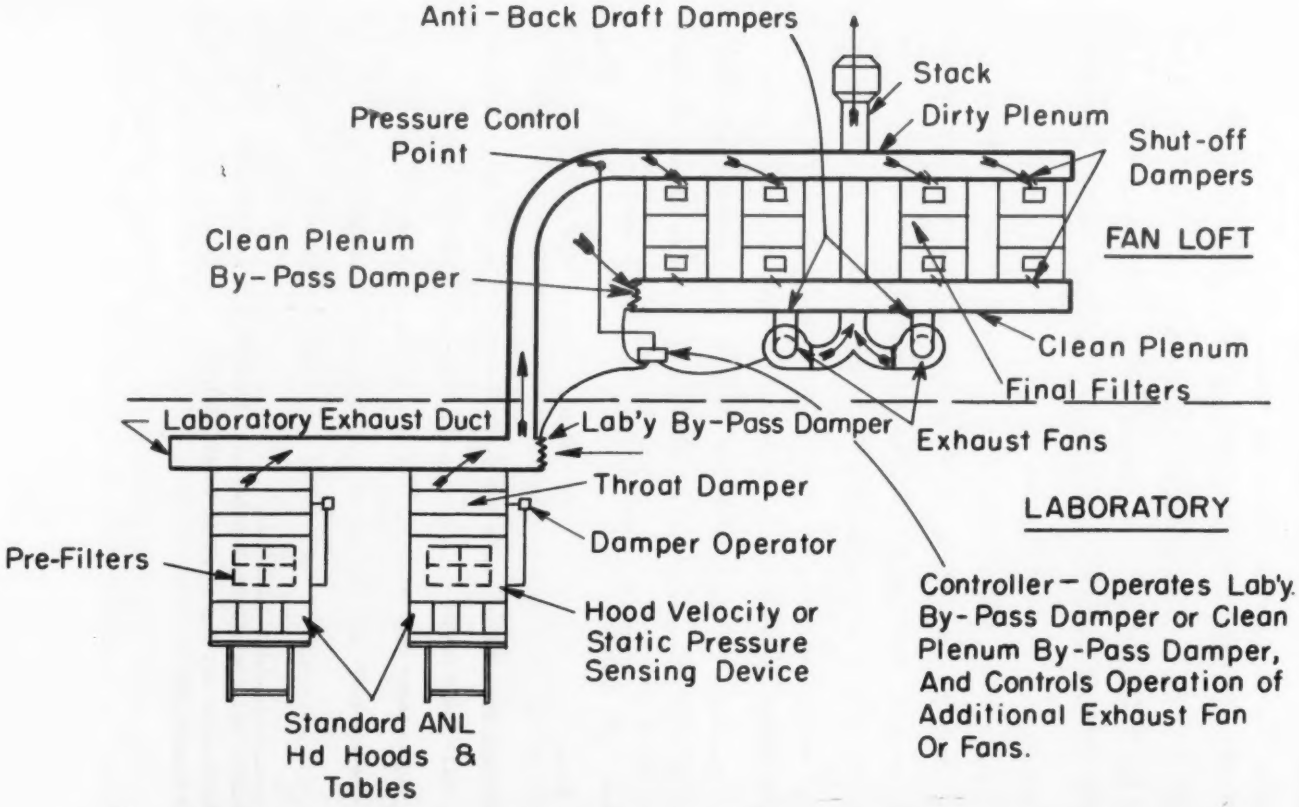


FIGURE 13.--Typical radioactive hood exhaust system with hood velocity control.

FIGURE 13.--Typical radioactive hood exhaust system with hood velocity control.

Costs

The total capital investment in the Argonne waste disposal system is $2,740,000. The annual operating charges are about $340,000 or roughly one per cent of the total Laboratory budget. These data are shown in Table 7. More detailed breakdowns of operating costs for solids and liquids giving unit costs are shown in Tables 8 and 9, respectively.

Table 7

CAPITAL AND OPERATING CHARGES
ARGONNE NATIONAL LABORATORY
WASTE DISPOSAL SYSTEM

Capital Cost of Facilities

	Capital Investment	Equipment	Total
Solid Waste System	$ 180,000	$ 23,000	$ 203,000
Liquid Waste System	1,865,000	215,000	2,080,000
Gaseous Waste System	695,000	---	695,000
Total	$2,740,000	$238,000	$2,978,000

Annual Operating Cost[a,c]

Solid Waste System	$ 90,000
Liquid Waste System	190,000
Gaseous Waste System	60,000
	$ 340,000[b]

[a] Includes all operations, maintenance, amortization and overhead
[b] Amortization and depreciation totals $88,500
[c] For Fiscal 1957

Table 8

OPERATING COST
SOLID WASTE DISPOSAL SYSTEM[a]

	Labor (man-hours)	Labor Cost[b]	Total Operating Cost[c]	Total Cost Including Laboratory Overhead[d]	Volume (cu ft)	Unit Cost[e]
Collection and storage Low Level	3,365	$ 9,774	$23,376	$30,218	23,919	$1.26
Collection and storage High Level	140	390	11,684	11,957	857	16.04
Concentration Low Level	374	1,107	3,021	3,796	10,208	0.37
Packaging Low Level	1,165	3,448	23,825	26,249	16,141	1.63
Shipment to Final Burial Low Level	441	1,212	16,284	17,132	16,100	1.06
Total	5,485	$15,931	$78,200	$89,352		

Unit Cost - Low Level $3.83/cu ft[e]
Unit Cost - High Level $16.04/cu ft

[a] Fiscal year, 1957

[b] Direct labor at $2.90/hr

[c] Includes Plant Services burden of 54 per cent of direct labor, material, depreciation and maintenance

[d] Includes Laboratory overhead of 70 per cent of direct labor

[e] Unit cost figures for low level waste are not additive directly because of volume change during concentration

[e] Unit cost figures for low level waste are not additive directly because of volume change during concentration

Table 9

OPERATING COST
LIQUID WASTE DISPOSAL SYSTEM[a]

	Labor (man-hours)	Labor Cost[b]	Total Operating Cost[c]	Total Cost Including Laboratory Overhead[d]	Volume (gal)	Unit Cost
Liquids Requiring Decontamination						
Individual Type Active Waste						
Collection, Temporary Storage and Transportation	1,069	$ 3,100	$ 7,000	$ 9,200	12,225	$0.75/gal
Treatment	1,256	4,200	14,600	17,500	31,830	0.55/gal
Total	2,325	$ 7,300	$21,600	$26,700		$1.30
Retention Tank Type Waste						
Collection, Temporary Storage and Transportation	654	$ 1,800	$ 6,200	$ 7,400	490,678	$0.015/gal
Treatment	5,024	16,700	56,900	68,600	199,790	0.34/gal
Total	5,678	$18,500	$63,100	$ 76,000		$0.36
Liquids Requiring noDecontamination	10,297	$23,800	$73,300	$ 90,000	86,749,918	$1.04/1000 gal
Grand Total		$49,600	$158,000	$192,700		

[a] Fiscal year, 1957
[b] Direct labor at $2.90/hr
[c] Includes Plant Services burden of 54 per cent of direct labor, material, depreciation, and maintenance
[d] Includes Laboratory overhead of 70 per cent of direct labor

Waste Management Operations at Brookhaven National Laboratory*

P. P. Cowan and L. Gemmell

Introduction

When Brookhaven National Laboratory was established shortly after World War II, it was realized that disposal of radioactive wastes would be a major problem and great care was exercised in planning suitable policies and procedures. The Laboratory is located at the geographical center of Long Island with farming and residential areas just beyond the boundaries of the 3600-acre site and with several villages at distances of 5 to 15 miles. Most of the radioactive waste is generated directly or indirectly by the large air-cooled research reactor. Fortunately, the BNL program has not included large-scale processing of fuel elements and thus waste disposal, while bothersome and expensive, has been reasonably manageable.

In discussing waste management operations at this laboratory, it will be convenient to consider gaseous, liquid and solid wastes separately. Sea disposal, the ultimate fate of most non-gaseous waste activity, will then be considered. The flow diagram shown in Figure 1 illustrates the major features and interrelationships of the waste control system.

Gaseous Waste Management

Although Brookhaven National Laboratory has a variety of hoods and ventilating systems associated with tracer or low level experimentation, these pose no particular problems as far as gaseous waste disposal is concerned. The only major sources of gaseous waste activity are the off-gases from the Hot Laboratory and the cooling air from the reactor. The off-gases from the Hot Laboratory are well taken care of by filters and scrubbers and by release through a pipe going up the center of the

*Prepared for the hearings on Industrial Radioactive Waste Disposal Problems to be held before the Special Sub-committee on Radiation of the Joint Committee on Atomic Energy on Jan. 28-Feb. 3, 1959.

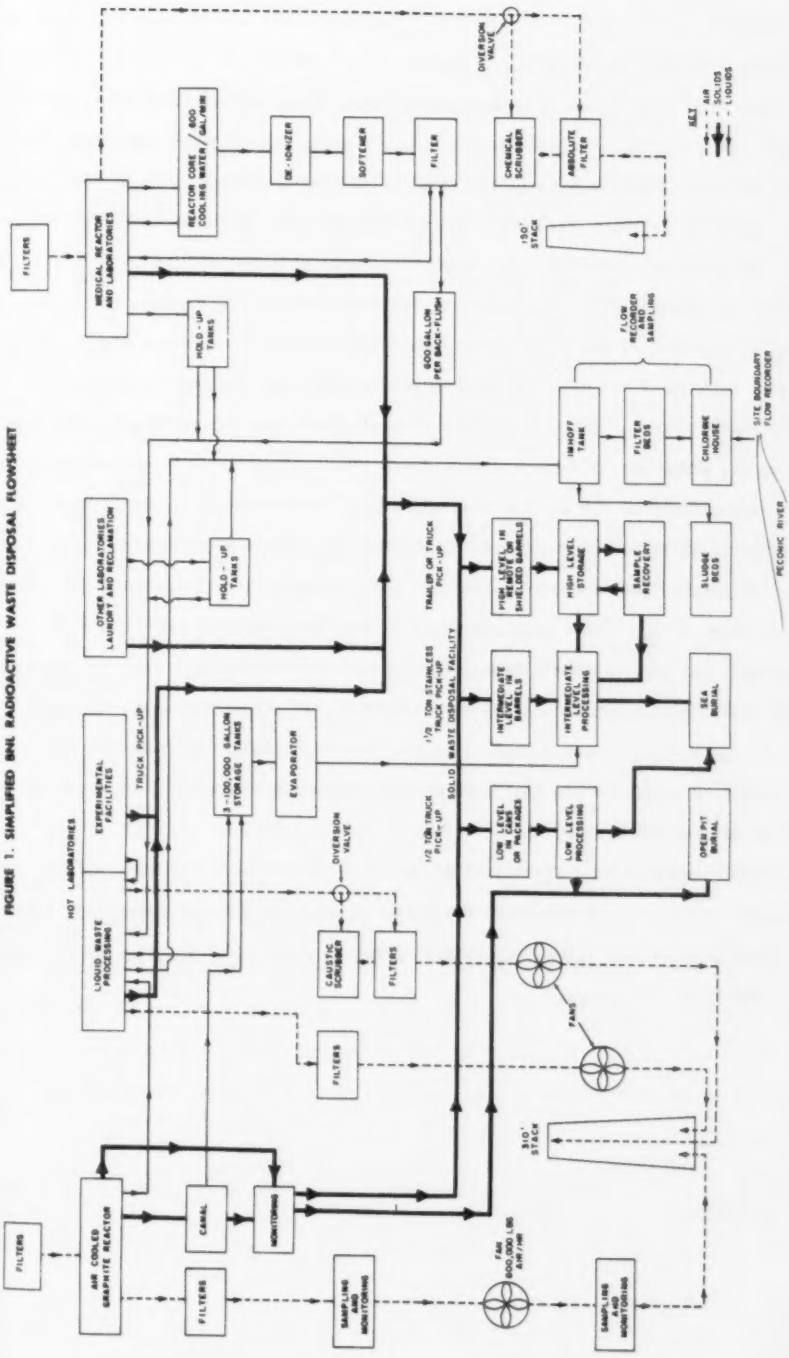


FIGURE 1. SIMPLIFIED BNL RADIOACTIVE WASTE DISPOSAL FLOWSHEET

310-foot stack provided for the pile cooling air. Thus, the major problem in gaseous waste disposal has been the reactor cooling air whose chief radioactive constituent is Argon-41.

Historical Background of the Argon Problem

The Brookhaven Research Reactor, until quite recently, released approximately 6,600 curies per day of Argon-41, which is the major radioactive constituent of air that has been briefly subjected to a flux of thermal neutrons. This rather considerable amount of activity was diluted in the 220,000 cubic foot per minute air flow and released from a 310-foot high stack whose base is about 30 feet above the immediate surroundings. When the reactor was first put into operation, the policy was to allow levels at the edge of the site that, when averaged over a week, would be no more than 1% of the limit prescribed for occupational exposure which, at that time, was 50 mr/day. Thus, the effect of the reactor was not permitted to exceed 3.5 mr/wk, which was about twice the natural gamma ray background level in this area.

With a limit as low as that just described, it was necessary, despite the very substantial dilution to be expected between the top of the stack and the edge of the site, to provide relatively sophisticated and expensive means for evaluating and controlling the exposure levels resulting from actual operation of the reactor. There were two separate but coordinated aspects of this program: first, a network of 15 area monitoring stations was established for routinely measuring the radiation levels; and second, a meteorological program was established for computing the levels from measured values of the pertinent weather parameters. Each of these programs will be briefly summarized.

Before going on to a discussion of the area monitoring and meteorological programs, it will be well to point out that, by joint AEC-BNL agreement, the limit for off-site radiation levels was raised to 10% of the occupational limits in 1953. As a result of this change,

the need for both of these programs was greatly reduced; indeed, a far less extensive effort would have sufficed had the 10% limit been initially specified. However, recent developments that will be discussed below have resulted in levels on-site that are much closer to the maximum permissible values and the need for area monitoring and meteorological evaluation has increased.

The Area Monitoring Program

Since Argon-41 is a noble gas that is not metabolized by the body, it poses merely an external radiation exposure problem. The half-life of the activity is 112 minutes and thus, although the total activity emitted amounted to 6,600 curies per day, no more than 730 curies was in existence at any one time.

In order to evaluate the radiation dosage caused by the Argon-41 from the reactor stack, a chain of 15 monitoring stations, located at distances up to 10 miles in various directions from the center of the site, was established[1]. These stations were equipped with an ionization chamber type monitor for gamma radiation,[2] several G.M. counters, a moving tape type of dust monitor[3] and a battery operated ratemeter. All of the data was recorded photographically every 6 minutes and the stations would operate for a full week before it was necessary to change the film in the recording camera. These stations were developed during 1948, at a time when relatively little suitable equipment was available commercially. The stations and their equipment have been described in several publications which are listed in the bibliography at the end of this paper.

Prior to start-up of the reactor, a full year of data was obtained to establish the naturally existing background radiation levels. The average values noted at that time are shown in Table I. The effect of the Argon from the stack at a given station was very small when averaged over a period of one year. For shorter periods of time the effect was quite substantial. In fact, it was estimated that peak levels of the

TABLE I

Background Radiation Data from All Stations
for Year Ending June 30, 1950

Type of Apparatus	Background Data	
	Average for Year	Range
A	30 c/m	26 - 36 c/m
B	30 c/m	25 - 36 c/m
C	25 c/m	20 - 28 c/m
E	34 c/m	29 - 37 c/m
V	1.6 mr/wk	1.3 - 2.0 mr/wk
F-α	2 c/m	1 - 3 c/m
F-$\beta\gamma$	26 c/m	22 - 34 c/m

A = approx. 35-50 mg/cm^2 glass-walled cylindrical G-M tube
B = approx. 35-50 mg/cm^2 glass-walled cylindrical G-M tube
C = Same as A and B but with 83 mil brass shield
E = Battery-operated ratemeter using same tube as above
V = Ionization chamber and dynamic condenser electrometer
F-α = Continuous dust monitor - Scintillation alpha counter
F-$\beta\gamma$ = Continuous dust monitor - Mica end-window G-M tube 2.5 - 4.0 mg/cm^2

order of 1 mr/hr could be experienced for a few minutes at on-site locations under certain meteorological conditions.

The Meteorological Program

A meteorological group was set up as a part of the Brookhaven Reactor project. A 420-foot high tower equipped with instrumentation for determining the wind and temperature structure as a function of height and time was provided. In addition, the group was equipped to plot weather maps and make forecasts. The tower was equipped with a smoke generator for releasing a test smoke at the same elevation as the top of the stack. The meteorologists developed analytical procedures for forecasting the dosage resulting from the pile effluent on the basis of diffusion theory, modified and standardized by experimental determinations of the dilution actually experienced by the test smoke effluent.[4,5] During the early years of reactor operation a daily prediction of the location of the 3.5 mr contour was prepared based on six days of actual data and one day of predicted data. Pile operations were regulated in terms of these computations and on two or three occasions it was found necessary to somewhat curtail operation of the reactor. Thus, the initial estimate of the possible seriousness of this problem was confirmed.

Recent Developments With Respect to Gaseous Waste Management

As noted above, the exposure limits applicable outside of the control area are now set at 10% of the occupational limits. However, the occupational limits for integrated exposures have been effectively reduced by a factor of 3 and must now average no more than 100 mr/wk from age 18 on. Thus, beyond the control area the prescribed limit is 10 mr/wk, or an average of about 0.06 mr/hr. However, it is permissible to average the exposure over a period of one year.

The BNL Research Reactor has recently been reloaded with enriched fuel and now releases approximately 2-1/2 times as much radioactive Argon, or about 16,100 curies

per day. In addition, the Medical Research Reactor will soon commence operation and, when running, will release an estimated 0.73 curies per hour of Argon-41 from a 150-foot stack located 1/2 mile in a southwesterly direction from the research reactor.

An analysis of the situation strongly indicates that the levels to be expected are comfortably within the prescribed limits, particularly in view of the longer averaging period now permitted. However, there is a renewed need for area monitoring measurements and for meteorological computations to quantitatively evaluate the new levels. It is interesting to note that the evaluation of these levels is made somewhat difficult by the increase in natural background caused by fallout from weapons tests. The background gamma ray activity observed at the stations has approximately doubled as compared with the values shown in Table I with very substantial variations above and below this figure, depending on the level of tropospheric fallout occurring at the moment. Thus, it is difficult to accurately evaluate disturbances due to the reactor in the region of 1 to 10 mr/wk.

At locations on site, levels considerably higher than those occurring at the site perimeter are encountered. However, a degree of control is exercised in all of these areas and they are, in general, occupied only 40 hours a week. Thus, while there must be an increased interest in on-site levels under the new circumstances, it is not anticipated that the average values will exceed the 10% limit applied beyond the controlled area.

Liquid Waste Management

Radioactive liquid wastes are generated at many locations on the BNL site. Most of the more active wastes originate at the reactor, hot laboratory or cyclotron buildings. Lower levels of activity are found in most of the other laboratories. The major features of the BNL liquid waste disposal system may be summarized as follows:

1. The underlying philosophy is to practice segregation at the source as much as is practical. Thus, containers are provided for collection of concentrated waste in all laboratories where such wastes are produced.

2. The Reactor-Hot Laboratory complex is provided with a separate liquid waste system so that release to the general laboratory system can be monitored and controlled.[6]

3. Local hold-up facilities are provided where there is the possibility that unacceptable amounts of activity might otherwise be released to the general laboratory sewerage system. Tanks are provided for hot wastes as a part of the Reactor-Hot Laboratory system.

4. An evaporation plant is provided to reduce the volume of liquid wastes that cannot be released to the general sewerage system.[7]

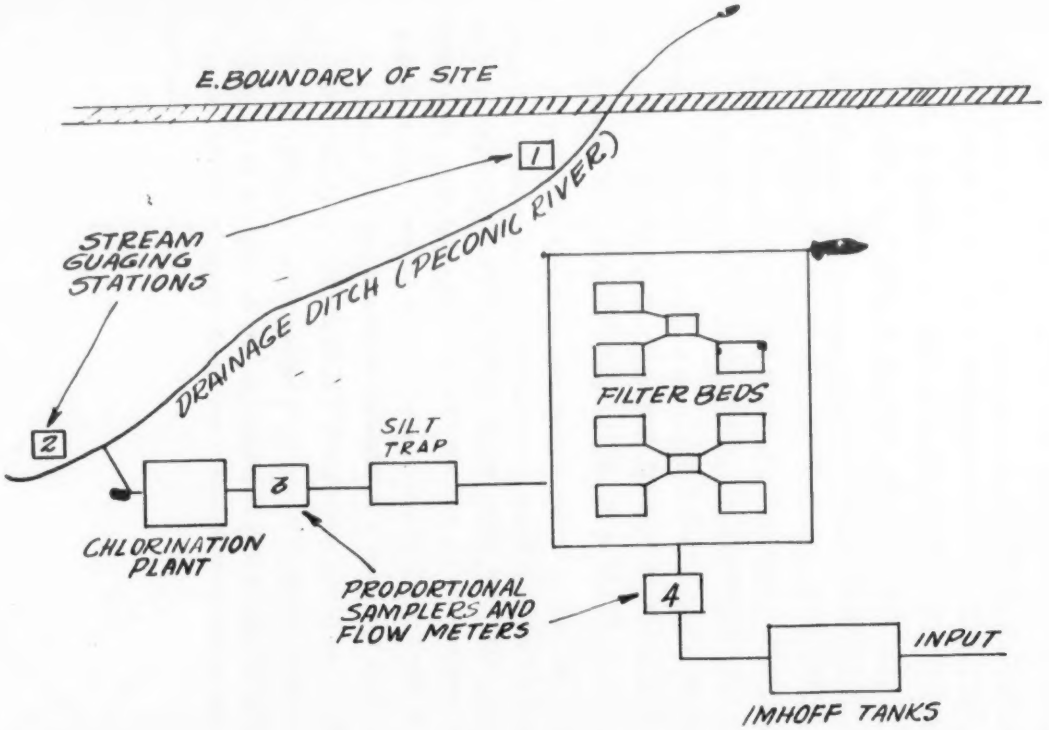
5. Active concentrate is used to make concrete which is packed in 55-gallon drums and disposed of at sea.

6. Monitoring arrangements are provided for the general laboratory sewerage system so that the agreed limit for the release of activity shall not be exceeded.

General Laboratory Sewerage System

Both the laboratory liquid wastes and the sanitary wastes are discharged to a sewerage system inherited from the days when this was a large Army camp. The effluent passes through an Imhoff tank which removes most of the solids and is then discharged to a large sand filter from which it is collected by an underlying tile field, is suitably chlorinated and is then discharged to a small stream. The stream is one of the headwaters of the Peconic River and has a negligible flow during dry seasons.

Figure 2 shows the monitoring arrangements provided for this central sewerage



B.N.L. Sewage Processing and Monitoring System

Figure 2.

system. Proportional samples are collected continuously at the point where the effluent is discharged to the filter beds and just prior to its discharge to the stream. These samples are analyzed daily for gross activity by simple evaporation and counting. The samplers are the Trebler variety in which samples proportional to the rate of flow are dipped up by a properly shaped rotating scoop from the pool on the upstream side of a weir. (see Figure 3) Flow meters are provided at the sampler locations and also in the stream above the discharge point and at the place where it leaves the site.

Early in the history of the laboratory it was decided by BNL-AEC to apply a limit of 3×10^{-6} μc/cc for gross activity at the point of discharge to the filter beds with the further stipulation that the total activity discharged per year should not exceed 1.5 curies. The policy of segregating activity at the source has been very successful thus far and the levels measured at the input to the sand filters have averaged only about 20% of the concentration limit mentioned above. Table II summarizes the data obtained for Fiscal Years 1957 and 1958. It will be noted that a large fraction of the activity released to the filter beds is filtered out or adsorbed and does not reach the stream. Thus, both the total activity and the concentration of activity actually leaving the Brookhaven site are extremely small. Tests have occasionally been made at downstream locations but the effect of the laboratory effluent has been thus far undetectable.

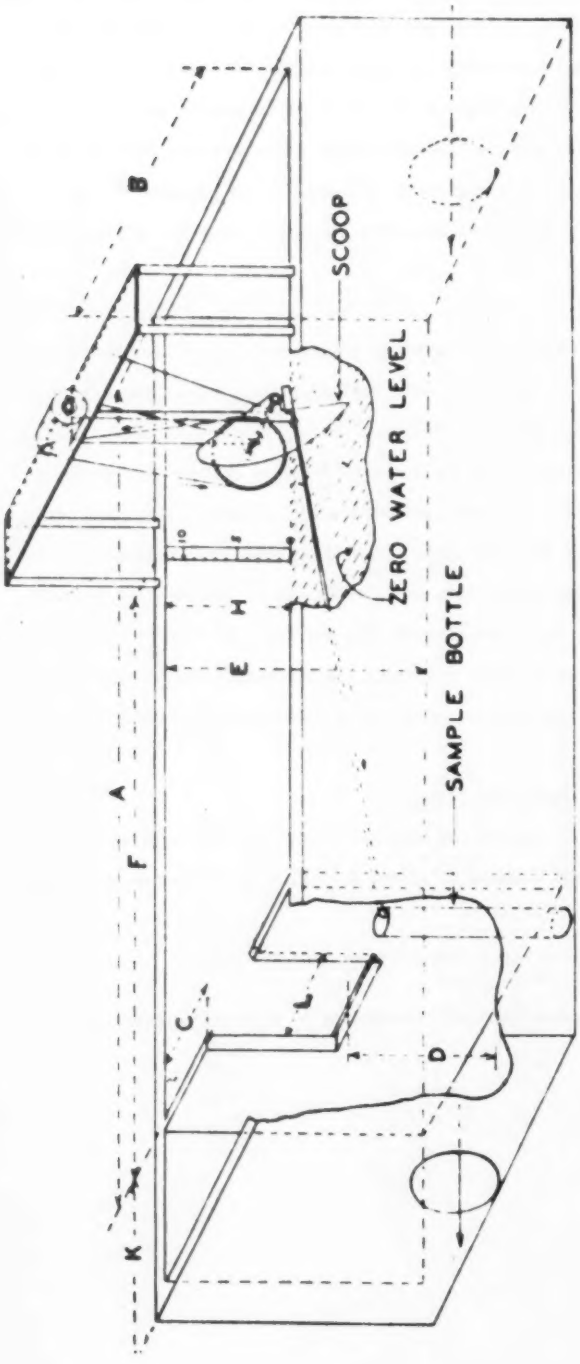
Figure 2.

Reactor-Hot Laboratory Liquid Waste Facilities

The Reactor-Hot Laboratory complex was originally provided with several piping systems for collecting different classes of liquid wastes, these being designated as follows:

A - Accountable metal waste containing uranium and other fissionable material.

B - High activity chemical waste containing no accountable material.



TREBLER SAMPLER SETTING

Figure 3.

TABLE II

Summary of Liquid Waste Data

	F.Y. 1958	F.Y. 1957
Input to filter beds	467,500 gal/day	427,500 gal/day
Output from filter beds	368,800 gal/day	332,000 gal/day
Net loss in filter beds	21.1 %	21.4 %
Stream above discharge point	934,400 gal/day	315,000 gal/day
Stream at site boundary	1,460,000 gal/day	778,500 gal/day
Rainfall	4.7 in/month	2.9 in/month
Activity concentration at input to filter beds	4.5×10^{-7} μc/cc	3.9×10^{-7} μc/cc
Activity concentration at output from filter beds	1.8×10^{-7} μc/cc	1.6×10^{-7} μc/cc
Reduction in activity concentration	60 %	58 %
Activity at input to filter beds	26.7 mc/month	17.5 mc/month
Activity at output from filter beds	8.2 mc/month	6.0 mc/month
Percent of permissible	21 %	14 %

Figures are averages as indicated for each fiscal year.

Figure 3.

C - Contaminated solvent waste containing volatile solvents or HCl.

D - Low activity chemical waste which does not require shielding.

F - Normally uncontaminated waste which is monitored for activity before disposal.

These piping systems are all brought together at a waste control center located in the Hot Laboratory. This center is provided with hold-up tanks where the wastes are stored until their activity can be determined and they can be discharged either to the tank farm or to the laboratory sewerage system. Hot wastes collected in containers elsewhere in the laboratory are brought to this location and dumped into the proper tank.

The tank farm is located 1/8 mile north of the Hot Laboratory waste control area. It contains three 100,000-gallon tanks and six 8,000-gallon tanks. An evaporation plant with a capacity of 10,000 gallons per shift week is located adjacent to the tank farm. This evaporator reduces the volume of the average waste by a factor of about 140.

Hold-up Facilities

In addition to the hold-up facilities associated with the Reactor-Hot Laboratory waste system, a variety of hold-up arrangements are provided elsewhere in the laboratory. The simplest arrangement is a single medium sized tank to which waste effluent that may become objectionably contaminated is discharged. When the tank is nearly full, the contents are analyzed and either discharged to the sewer or, when necessary, pumped out and transported to the Hot Laboratory by tank truck. For laboratories with a larger volume of flow, larger tanks are provided in pairs. When one tank is nearly full, the flow is diverted to the second tank while an analysis is made of the contents in the first tank and arrangements are made for their proper disposition. Analysis of the activity in a hold-up tank of this type is facilitated by use of a sampler such as is shown in Figure 4.

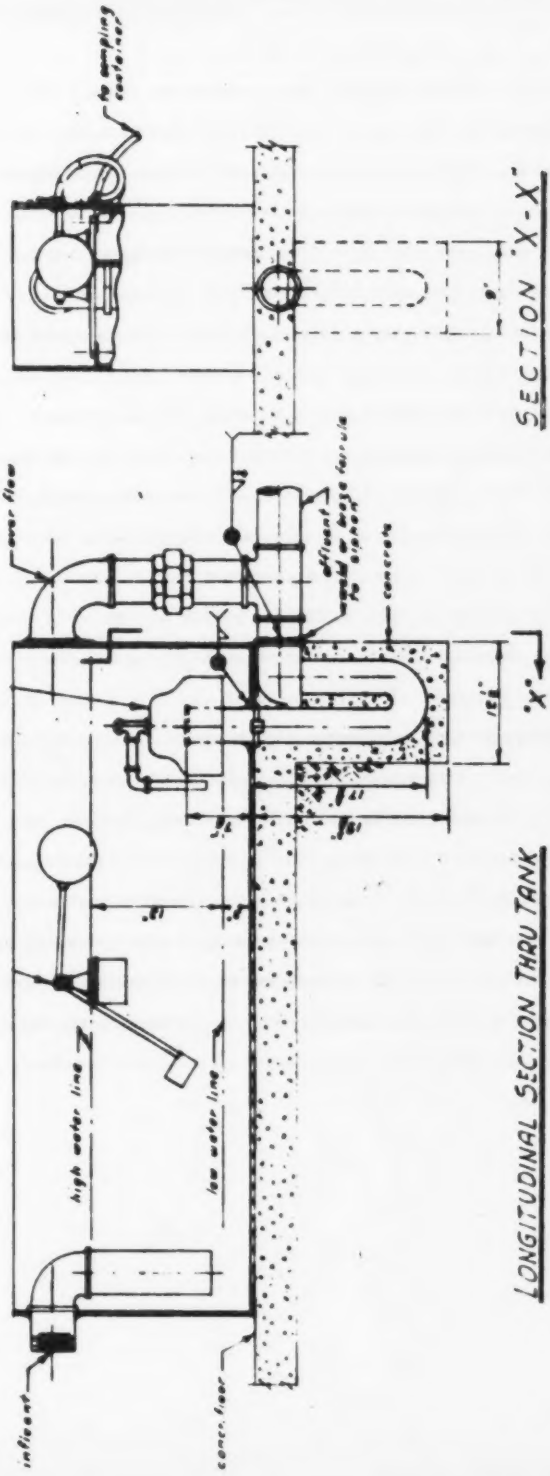


FIGURE 4.--Proportional sampler.

Solid Waste Management

Solid wastes may be produced wherever experimenters are working with radioactive materials. Areas such as the Reactor and Hot Lab generate considerable quantities of fission product waste. Activated objects from the Reactor, including experimental equipment, thermocouples, pile structural materials, graphite, irradiated samples, furnaces, autoclaves, etc., are used in various experiments and then become waste. Cleaning mops, rags, swipes, etc. used for decontamination become waste. Left-over isotopes and excreta from animals treated with radioactive isotopes must be disposed of. The liquid slurry from the evaporator mentioned above must be solidified by some additive such as cement so that it can be treated as solid waste.

At BNL these wastes are handled in one of three ways depending on the amounts and types of radioactivity involved. If the levels of radioactivity are low, a considerable quantity can be thrown away locally on the dump area. Wastes that are too active for local burial will ordinarily be packaged for sea burial. The high level wastes that are "too hot" to package for sea burial are stored in underground concrete-lined, waterproofed pits for decay and possible subsequent disposal.

Policies Governing Local Disposal

Initially, the Laboratory policy did not permit the burial of any radioactive solid wastes. In 1953 the policy was restudied and revised to allow annual local burial of as much as 10 curies of activity with half-life less than 100 days, 1 curie with half-life between 100 days and 5 years and 1/10 curie with half-life longer than 5 years. The amounts of Pu^{239} or Ra^{226} disposed of in this manner were limited to 1 millicurie each. Thus, most of the short-lived isotopes used by the biologists and medical personnel can be handled by storage and decay, with final disposal in the burial ground. The 100 millicurie ceiling for long-lived activity permits certain bulky equipment and containers slightly contaminated with uranium, thorium, etc. to be buried locally.

Methods of Processing Solid Wastes

As previously mentioned, segregation at the source is perhaps the best way to control the handling of radioactive waste. Wastes that are made in laboratories are wrapped in paper so that there will be no spillage while they are being transported to the waste processing area. These packages are identified with a tag or label showing the isotope within, the radiation level at contact with the package and the experimenter's estimate of the curie content of the package. With this information, the waste processors can decide the fate of the contents of the package.

At the Reactor, the wastes are segregated into several categories. At the various refuse stations around the building there are always two 20-gallon metal containers together. One is labeled for "Clean Refuse Only" and the other for "Radioactive Waste". The "clean waste" is superficially monitored to be sure that no appreciable amount of radioactive waste gets into the clean waste by mistake. It then goes to the dumping area where it is burned and buried. The waste that is put in the can labeled for "Radioactive Waste" is transported to the waste processing area where it is monitored. Those items that are essentially non-radioactive are sent on to the dump area and those items that must be packaged for sea burial go into 55-gallon drums.

Low level packaged wastes from laboratories and the wastes placed in cans are transported to the waste area by an ordinary pick-up truck without restrictions on the driver who is required to wear a film badge but is not expected to make radiation measurements. The higher level wastes are picked up by a member of the waste processing group in a truck specially designated for this purpose. In general, most of this type of waste will be packaged for sea burial. See flow sheet, Figure 1.

A third type of much more radioactive waste consists of large items that have been discharged from experimental holes in the reactor. These experimental equipments are usually pulled by remote handling equipment into long wooden boxes, mounted

on a trailer and trailed behind a truck to the waste processing area. These items are usually too "hot" to handle so they are dropped into an underground storage pit as previously mentioned. After months or years of decay, some of these objects may be processed for sea disposal. If there are valuable samples inside these highly activated objects, the experimenter may elect to allow sufficient cooling time for safe working conditions or he may recover his samples by the use of a shielding wall and remote handling equipment.

Packaging for Sea Disposal

In the past, Brookhaven has used 55-gallon drums as the standard package for sea disposal. This choice was dictated by four factors:

1. The Navy LST that has hauled the waste to sea has preferred this size and shape.
2. Salvage drums have been available at no cost other than transportation to the Lab site.
3. 55-gallon drums can be handled by ordinary drum handling equipment.
4. This size container holds a reasonable amount of waste.

For sea burial it is desirable that all wastes be in solid form. As stated before, the slurry from the evaporator is mixed with cement in a batch-type mixer and poured into drums to harden. Glass vials, etc., are broken and bulky metal objects are crushed to leave as few voids in the drum as possible. Acids are neutralized and solidified. No liquids are permitted because of the difficulties that might ensue should a drum of radioactive liquid break open and spill along the highway.

The radiation levels at the outside of the drum must be held down since overexposure of personnel packaging and handling the drums during disposal must be avoided. Frequently, concrete shields must be poured into the drums prior to packaging the material, as shown in Figure 5. The concrete is poured around heavy cardboard forms

FIGURE 5.--Shielded drum.

and the shields may vary in thickness from as little as 2 inches up to 10 inches. The waste is placed in the cylindrical hole and then concrete is poured in on top of it to fill up the drum.

Combustibles such as paper, rags, mops, etc. that are too active to be placed on the dump are compressed into bales approximately 1 foot square and 20 inches high. Each bale is placed in a drum and concrete is poured around it. This size of bale allows for enough concrete to insure that the drum will sink. Each drum is weighed, however, to make sure that it is heavy enough. Incineration of combustibles was tried for awhile but has been abandoned here as well as at most of the ASC laboratories because of the large costs involved in cleaning up the smoke.

After the drums have solidified and are ready for shipment, they are monitored for radiation and a record is kept of each drum by number. The record indicates what is in each drum, the radiation level at the outside and an estimate of its curie content. The drums are then stored on the ground upside down so that rain water cannot seep into them. They are staged on the ground in the same general pattern that they will occupy as they go aboard the trucks that will transport them to the shipping wharf, 48 drums to a load. The least active drums are placed in the outside rows so as to offer some extra shielding for the more active drums in the inside rows.

After the drums are staged, they are all color-coded to indicate the radiation level of each. This enables those individuals handling them during the trucking, loading and shipping operations to tell which drums are the most hazardous. Drums awaiting shipment are stored in an area secured by an eight-foot anti-personnel fence that is kept locked at all times when there is no one in attendance.

Sea Disposal

Since there is no suitable designated approved burial site for radioactive materials closer than Oak Ridge, Tennessee, it is easier and more economical for us

to ship this waste to sea. The Navy has been disposing of surplus Naval Ordinance materials collected at Earle, N.J., by using an LST. The New York Operations Office of the AEC has made special arrangements with the Navy to pick up full ship load quantities of the BNL waste from the wharf at Floyd Bennett Field in Brooklyn. This arrangement has been carried on successfully once each year since 1954.

When a shipload of approximately 700 drums has accumulated at BNL, a shipping schedule is set up. A commercial trucker who can furnish a sufficient number of flat bed trucks at one time is hired to haul the drums to the wharf. The drums are loaded aboard the trucks by Brookhaven personnel and Brookhaven equipment. The trucks proceed to the wharf in convoys with an escort car for each convoy. The escort provides extra radiation advice and protection in case of an accident. Brookhaven men and equipment off-load the trucks onto the wharf.

On the third day of this transportation schedule, the LST arrives at Floyd Bennett Field and begins to load from the wharf to the top deck; see Figure 6. The crane is furnished by an outside contractor and the ship's crew handle the placement of the slings around the barrels on the wharf and placing of drums in the bins. After the ship is loaded she sails promptly. The ship's crew is moved from their quarters amidships to fore and aft bunk space because of the radiation levels in the center part of the ship. Each crew member, as well as any other person associated with this operation, wears a film badge at all times. Self-reading dosimeters and other ratemeter types of instruments are provided so as to evaluate radiation levels and keep exposures at a minimum. At least two Health Physicists supervise the entire operation from the standpoint of radiation protection.

The ship sails to a pre-designated dumping area located approximately 250 miles south-southeast of Brooklyn. At this location the water is about 2100 fathoms deep. The sailing time out is ordinarily about 24 hours, with only 60 minutes required to

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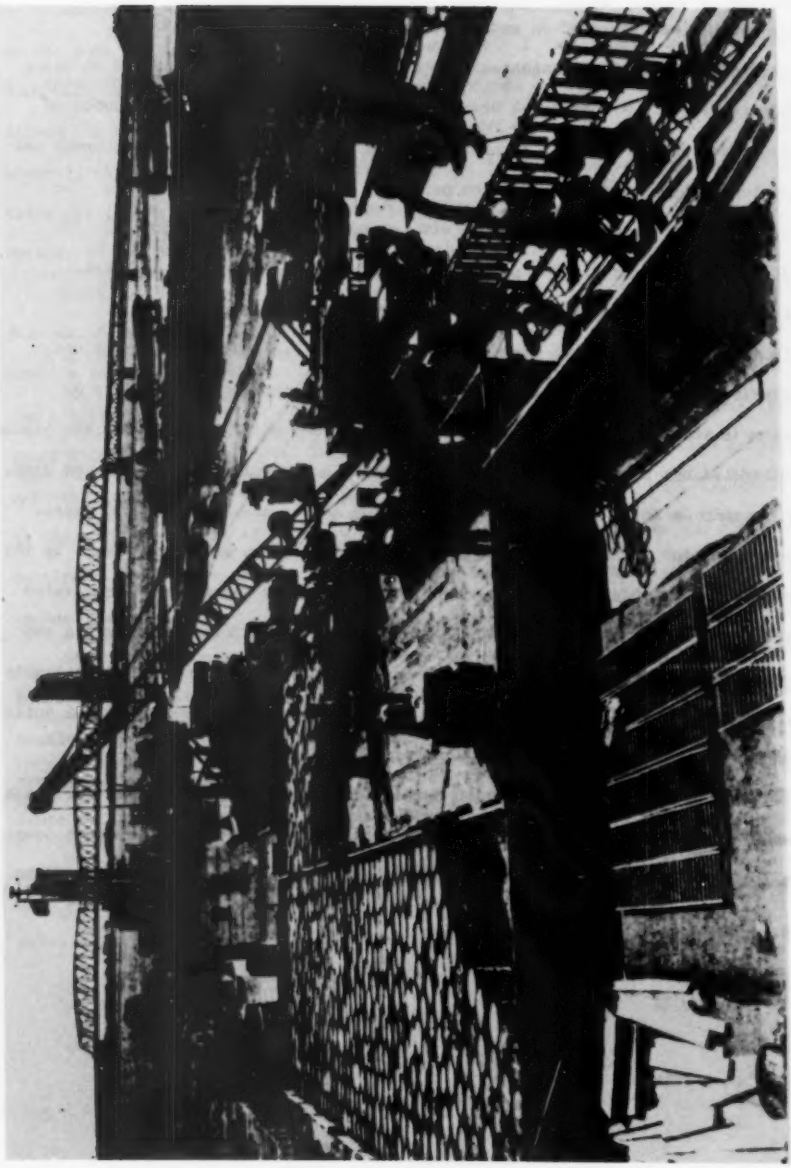

FIGURE 6.—Loading drums aboard L.S.T.

roll the drums off the ship. On the way back, the ship is surveyed for radioactive contamination. If decontamination is required, it is done under the supervision of the BNL Health Physicists.

During these sea disposal operations there have been no overexposures to the crew or others involved and the amount of contamination on the ship has been negligible. The pertinent data on past shipments is listed for the record.

1954	472 drums	233 curies
1955	658 "	597 "
1956	638 "	660 "
1957	704 "	700 "
1958	692 "	1100 "

In 1959 there will probably be two shipments totalling about 1400 drums containing approximately 2000 curies.

Those persons packaging and disposing of the BNL waste at sea have been careful to see that all drums were heavy enough to sink. There is undoubtedly some drum failure by implosion at these ocean depths, particularly in the case of those drums containing low-level combustible wastes. However, the amounts of activity disposed of are comparatively small so that leaching of the contents by sea water is not a matter for concern.

It has recently been announced that the Navy will transfer its sea disposal activities, including that carried out for BNL, to the Military Sea Transport Service. Some preliminary discussions have been held, but many details must be worked out before a satisfactory new procedure for dumping radioactive wastes from Brookhaven at sea can be set up. There is an urgent need for disposal of waste containers considerably larger and heavier than the 55-gallon drums now in use. It is hoped that the new arrangement will facilitate the accomplishment of this objective while still affording a drum disposal service as satisfactory as that presently provided by the Navy.

FIGURE 6.—Loading drums aboard L.S.T.

Only a rough analysis has been made of the costs of sea disposal of radioactive wastes as now practiced at BNL. The following figures are intended to include all costs exclusive of the cost to the Navy of transporting the waste to sea and dumping it. The labor charge covers monitoring, collecting, sorting, processing, record keeping, loading on trucks, etc.

labor.	$ 13.00 per drum
materials.	7.00 " "
trucking	2.00 " "
direct costs	$ 22.00 per drum
insurance.	1.00 " "
overhead	11.00 " "
capital depreciation .	3.00 " "
	$ 37.00 per drum

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General Electric Company Knolls Atomic Power Laboratory Schenectady, New York

RADIOACTIVE WASTE MANAGEMENT AT THE KNOLLS ATOMIC POWER LABORATORY

The Knolls Atomic Power Laboratory has been operated by the General Electric Company for the Atomic Energy Commission since 1946 under contract W31-109-Eng 52. Currently, the principal task of the Laboratory is to support the Naval Reactors Program of the Commission in the development of atomic power reactors for naval propulsion -- specifically, for submarines and destroyers. This task includes design, construction, and prototype operation of nuclear power reactors.

Location and Size of the Laboratory and Major Facilities

The Knolls Atomic Power Laboratory consists of two sites, the Knolls Site and the West Milton Site. The former is located 5 miles east of Schenectady, New York, on the south bank of the Mohawk River. It consists of approximately 170 acres, upon which are located administrative buildings; chemistry, physics, metallurgical, engineering, and radioactive materials laboratories; critical assembly buildings; machine shops; decontamination laundry; radioactive waste storage and processing facilities; nuclear fuel storage and assembly buildings.

The West Milton Site is located approximately 18 miles north of Schenectady, in Saratoga County. It occupies approximately 4,000 acres, and principal facilities include the Triton prototype reactor (S3G); equipment service building, fuel service building, and waste treatment facility in addition to administration, and utility service buildings. There is also a containment "sphere" in which the destroyer prototype (D1G) is being constructed. The D1G program will not create any radioactive waste for another year or so.

Operations Producing Wastes

Radioactive wastes at KAPL are produced in solid, liquid, and gaseous form by the technical laboratories, shops, decontamination laundry, power reactor operation, and other operating facilities which employ radioactive materials. Solid radioactive wastes include paper(that has been used as wipes, lab. bench covers,); cloth (clothing, rags); rubber and plastic (gloves, tubing, bottles, tile); glass (vials, bottles, syringes, tubing, etc.); wood (air filters, flooring); metal (tubing, pipes, tanks, equipment, irradiated fuel elements, etc.); and chemical (evaporator slurries, ion exchange resins). Liquid wastes consist of chemical laboratory sink drainage, wash and rinse waters from the decontamination laundry; equipment decontamination solutions, evaporator condensates; machine shop oil coolants, critical assembly and reactor water coolant, etc. Gaseous wastes are mostly air exhaust from laboratory hoods, hot cell ventilation, critical assembly room ventilation, machine shop ventilation, etc.

Radioactive Waste Considerations in Design

In general, requirements for the treatment and disposal of radioactive wastes are provided in the design of the operating facilities.[1,2] For example, demineralizers, retention tanks and evaporators for liquid wastes; surface and subsurface storage areas for solid wastes; and air cleaning systems (filters, cyclone separators, and scrubbers) for ventilated air are incorporated in the design stage. Occasionally, additional requirements develop after operating experience is gained and equipment and facilities are added. At KAPL, baling of compressible solid wastes to reduce shipping volumes and surface storage of wastes for decay purposes, were added after operations were begun to improve the economics of waste management.

Environmental Considerations in Radioactive Waste Management

Disposal of radioactive wastes into the environment has been limited to low level liquid wastes released into the Mohawk River at the Knolls Site and into Glowegee Creek which in turn flows into the Kayaderossaras Creek at the West Milton Site; and low level contaminated air released into the atmosphere at both sites. Solid wastes are stored temporarily and shipped to Oak Ridge National Laboratory for burial. Radioactive liquid wastes that will later be released into surface streams are first collected in retention tanks and monitored to determine the level of radioactivity. Ventilated air from laboratories, and operating areas employing radioactive materials, is cleaned by means of appropriate air cleaning systems and then monitored.[3,4] At the Knolls Site the Mohawk River and the atmosphere are used to dilute radioactivity released as liquid and gaseous wastes. Where dependence is placed upon environmental dilution of radioactive wastes, technical studies have been made of the dilution potential of the environment from both a short- and long-range basis, to assure that releases are within recommended levels.[5-11]

Environmental Monitoring

To assure that radioactive waste releases to the environment are under adequate operational control, an environmental monitoring program is maintained at both the Knolls and West Milton Sites. This program includes the determination of radioactivity levels in the surface streams, ground water, on vegetation and in the atmosphere. Monitoring of the Mohawk River includes the determination of radioactivity levels in the River water, mud, and aquatic life. Environmental monitoring findings are reported periodically in laboratory reports.[12-38]

Environmental Standards

The recommendation of the National Committee on Radiation Protection (through its published Handbooks)[39,40] on radiation exposure are followed in relation to environmental standards. In general, these standards are specified by the Atomic Energy Commission[41,42] and the New York State Health Department[43]. For environmental exposure, the one-tenth of the maximum permissible continuous exposure limit has been applied. In the case of the Mohawk River, the classification of this stream by the New York State Water Pollution Control Board has been recognized and radioactive waste disposal practices have been factored in to meet the usage classification requirements of the River.[44]

Administrative Aspects

Historically, radioactive waste disposal practices at the Laboratory have had the approval of the Schenectady Naval Reactors Operations Office of the Atomic Energy Commission. That office also had access to the assistance of the Sanitary Engineering staff associated with the Commission's Division of Reactor Development in reviewing KAPL radioactive waste disposal practices.

Early in the operating history of the Laboratory, both the Commission and the General Electric Company recognized a need for contact with the New York State Health Department and the city of Schenectady to discuss Laboratory activities as they specifically related to the Mohawk River and the discharge of waste into it. To accomplish this, the Manager of the Schenectady Naval Reactors Operations Office with the support of the Division of Reactor Development in Washington was instrumental in establishing the Mohawk River Advisory Committee[45].* The first meeting of this Committee was held at KAPL on October 30, 1952. Since then five additional meetings have been held to discuss KAPL liquid radioactive waste disposal practices at both Laboratory sites, and certain Research and

*The Mohawk River Advisory Committee membership includes: The Commissioner of Health, the Deputy Commissioner of Health, and the Director of Environmental Sanitation - all of the Department of Health, New York State; the Executive Secretary of N. Y. State Pollution Control Board, and a secretary who is a former City Manager of the City of Schenectady and who is now an engineering consultant on sanitation.

Development studies performed by the U. S. Geological Survey. In all of its activities this Committee has provided advice and counsel on matters relating to the release of radioactive waste to surface streams. This Committee has provided valuable liaison with state and municipal authorities interested in these matters.

KAPL radioactive waste practices also have been reviewed periodically at the Sanitary Engineering and Air Cleaning Conferences[3,4,46] sponsored by the Commission.

NATURE AND CHARACTERISTICS OF KAPL RADIOACTIVE WASTE

Radioelements in Waste

The principal radioelements found in KAPL wastes are natural and highly enriched uranium, plutonium, fission products, and reactor corrosion products, e.g., radioactive iron and manganese, etc. These radioelements are present in various degrees in solid, liquid, and gaseous waste effluents. Radioactive noble gases, such as Xe-133, 135, Kr-85, are generally present whenever irradiated nuclear fuel is dissolved. Argon-41 is present if air is activated by neutrons in or near the reactor.

Volume and Activity Level

Annual liquid waste production is estimated at 2,700,000 gallons. A major portion of this waste, approximately 1,800,000 gallons, containing approximately 25 curies, is released into the Mohawk River after in-plant dilution. Another large volume, approximately 900,000 gallons containing approximately 10 millicuries after in-plant dilution, is released into the Glowegee Creek at the West Milton Site. A small volume of about 200 gallons containing 10,000 curies is stored in underground stainless steel tanks housed in reinforced concrete cells.

Currently, it is estimated that the Laboratory produces approximately 40,000 cubic feet of solid wastes (including slurry from evaporators). Of this volume approximately 38,500 cubic feet is only slightly radioactive containing

approximately one curie. The remaining waste contains approximately 10,000 curies and, for economic reasons, is stored on the site until radioactive decay makes shipment to a burial ground economically feasible.

Annual air effluent released to the Knolls Site environs is estimated at approximately 100 billion cubic feet and contains an estimated 1.5 curies. Projected annual air effluent release to the West Milton Site environs is estimated at 4.6 billion cubic feet with radioactivity estimated at less than 1 millicurie.

Operational Controls

The release of liquid wastes into the Mohawk River is regulated by controlling the discharge rate of the effluent and regulating the concentration of fission products in the collection tanks. Wastes exceeding 5×10^{-2} μc/ml (.05 microcuries per millilitre) are sent to evaporators. For waste below this activity level it has been determined that river dilution will bring the concentration of all radioisotopes present to less than 1/10 the maximum permissible concentration specified in NBS Handbook 52. Radioactivity hazards from this material is very slight; isotopic analysis of liquid wastes generated at the Knolls Site show that Strontium-90 is the principal component which has to be considered in hazard control.

The release of liquid waste effluents at the West Milton Site into Glowegee Creek is operationally controlled by limiting the level of radioactivity to a concentration of $2 \times 10^{-6*}$ μc/ml. With this control limit it has been determined that concentrations of all radio nuclides in the effluent are less than 1/10 maximum permissible concentration specified in NBS Handbook 52. This operating control limit is based on the Sr-90 contribution of 3.7% to the total activity, the Sr-90 content being more significant biologically than the reactor corrosion products. The Glowegee Creek is not relied upon to dilute radioactive wastes.

Generally, the only operational control on radioactivity released into the atmosphere from Laboratory stacks is the air cleaning system provided. However, in some areas at both the Knolls and West Milton Sites, certain ventilation systems

*This limit will vary with isotopic content of waste as determined by radioanalysis.

can be automatically shut down by a signal resulting from an emergency radiation situation in a critical assembly or power reactor.

CURRENT RADIOACTIVE WASTE MANAGEMENT OPERATIONS AND COSTS

AIR-BORNE WASTES

Air-cleaning equipment is a fundamental part of exhaust systems servicing areas and processes capable of generating air-borne radioactive wastes. High efficiency filters are used to clean ventilation air in the exhaust systems of radioactive particles as small as sub micron in size. In addition, Laboratory facilities that are used for examining irradiated fuel are equipped with caustic scrubbers for removing volatile radioactive iodine and ruthenium released during dissolving. Radioactive noble gases (Xenon and Krypton) produced by the fission process are released to the environs at controlled rates following collection and analysis.

LIQUID WASTE

Liquid wastes are accomodated in drainage systems that lead to collection tanks. Liquids with low levels of radioactivity, not exceeding operational control limits, are released to the Mohawk River from the Knolls Site and to the Glowegee Creek from the West Milton Site. Liquids with high levels of radioactivity that exceed operational control limits for release to the environment are processed by demineralization or evaporation. The radioactivity in resin and evaporator slurry is disposed as solid waste.

SOLID WASTE

Low level radioactive solid wastes, e.g., paper and cloth wipers, protective clothing, air filters, floor sweeping, scrap metal, etc., are collected in waste cans or similar containers, baled if compressible, packaged in ICC specification boxes and drums, and transported by rail to the Oak Ridge National Laboratory (ORNL) for burial.

SOLID WASTE (Continued)

Intermediate level radioactive solid wastes, e.g., air filters, benches, hoods, and other miscellaneous equipment, are packaged in ICC specification boxes and drums by using remote handling equipment. They are then stored in a shielded storage area until radioactive decay reduces gamma radiation from the container to a level which permits efficient and economic shipment by rail to ORNL.

Highly radioactive solid wastes, such as irradiated reactor material samples in the Radioactive Materials Laboratory at the Knolls Site, and resin demineralizers in the S3G coolant purification system at the West Milton Site, are loaded remotely into shielded containers and stored in a shielded area.

PROCEDURES AND FACILITIES -- KNOLLS

The following is a description of the facilities and operational procedures for the disposition of KAPL radioactive wastes at the Knolls Site. For reference, see Figure 1, noting carefully the numerical sequence there.

Solid Wastes

High Level, Canning

The canned wastes are generated in the Radioactive Materials Laboratory cells (RML, Bldg. E-4). These wastes are remotely placed in aluminum liners while they are in the cells. The liners, 4 1/4" OD; 3 3/4" ID and 44" long, have a welded bottom and threaded cap. After the container is loaded, it is placed in a lead shielding cask, removed from the cell, and moved to a temporary storage area prior to subsurface storage.

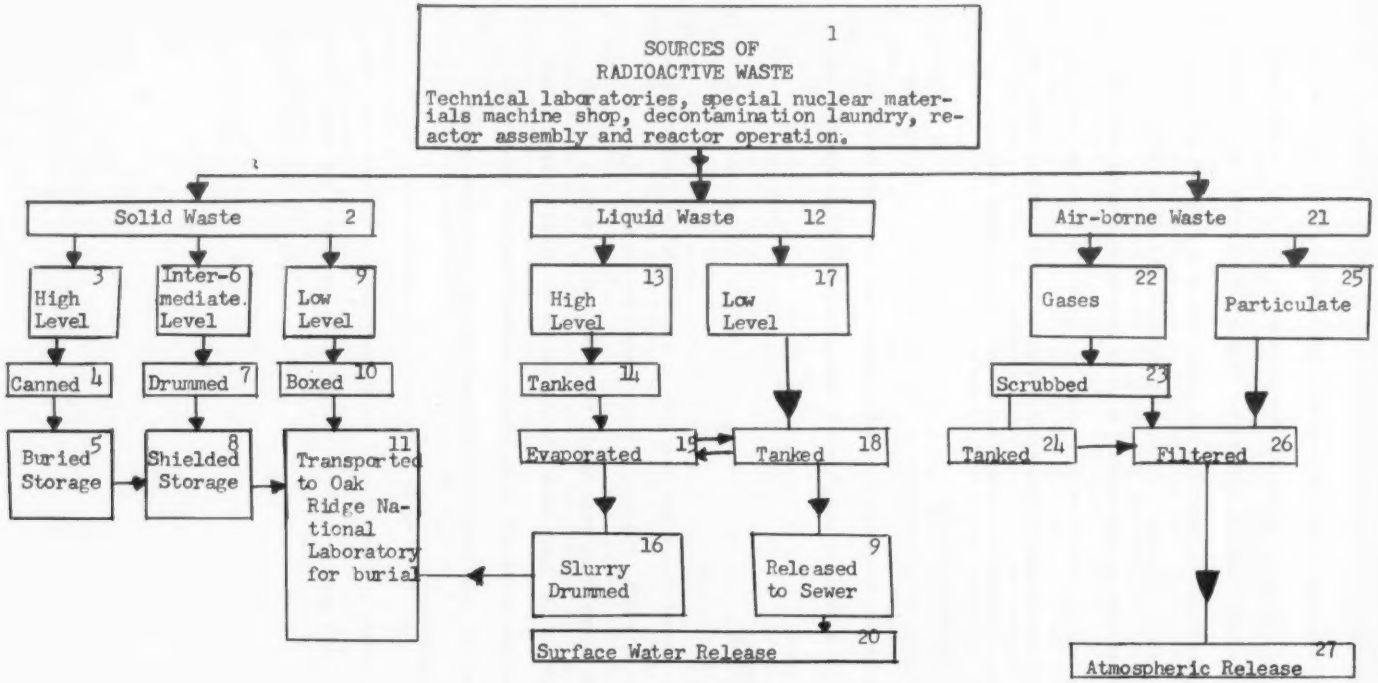


Figure I

Radioactive Waste Disposal Diagram of the
Knolls Atomic Power Laboratory (Knolls Site and West Milton Site)

Knolls Atomic Power Laboratory (Knolls Site and West Milton Site)

High-Level Sub-surface Storage

The buried storage area (M-4) consists of a reinforced concrete mass 14 feet in depth, with 250 sections of pipe installed in its walls and foundation. The high level waste and cask (Figure 1, Box No. 4) is moved into the area, and a portable gantry crane positions the cask over appropriate pipe section. The liner is remotely transferred from the cask to a pipe section for storage and remains there until sufficient radioactive decay permits its storage in a more economical above-ground storage area or shipment to ORNL for disposition. No such shipment has occurred to date.

Intermediate Level, Drumming

The drummed storage waste is mostly generated in the Radioactive Materials Laboratory. These wastes are stored in 55 gallon carbon steel drums. The surface dose rate is 25 r per hour or less; these containers may be handled safely with fork trucks or hand trucks. These waste containers are then moved to an above-grade shielded storage area. (Figure 1, Box 8)

Intermediate Level, Temporary Shielded Storage

The above-grade shielded storage area is an uncovered, two-room labyrinth of reinforced concrete. It has walls two feet, eight inches thick on three sides and a one-foot thick retaining wall against an earth bank on the lowest side. A one-foot thick partition separates the two rooms. The enclosed space has a one-foot thick reinforced concrete floor with a floor drain that is connected to the storm sewer system. Drummed wastes are stored here to permit decay of radioactivity to a level which permits more efficient handling, packaging, and shipment.

Low Level, Boxing

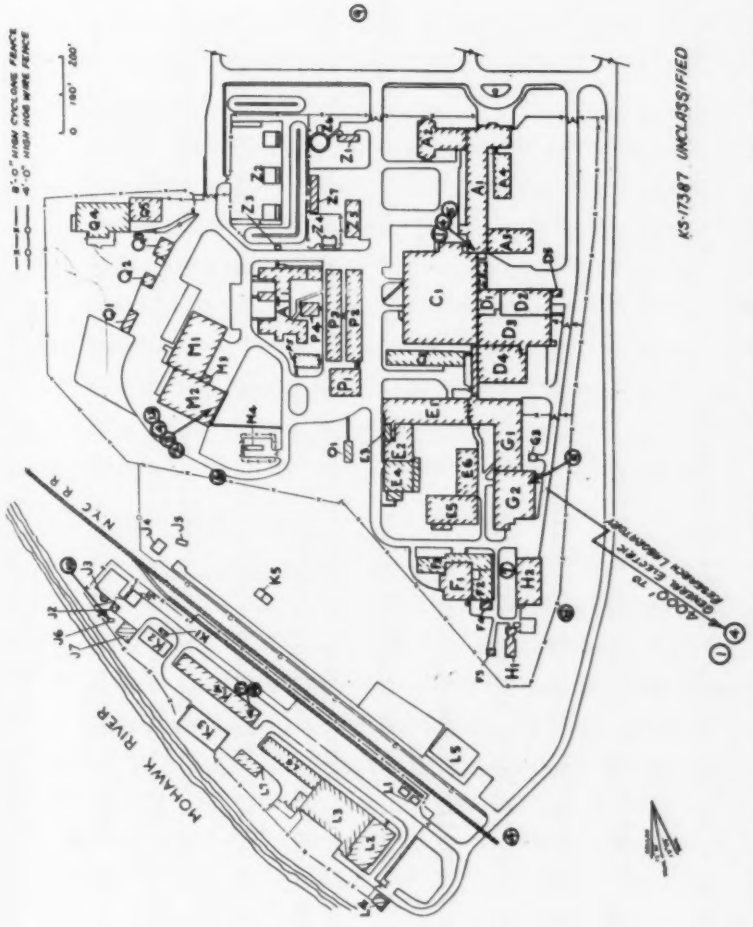
Low level wastes are collected in separate, 20-gallon containers that are lined with waterproof paper. The wastes are classified as compressible (balable) and non-compressible waste. The bagged compressible waste is deposited in a Dempster-Dumpster, and baled and boxed in ICC approved shipping containers. The non-compressible waste is put in ICC-specified boxes for transport to a concrete block building, L-7 (Figure 2). The baled containers are accumulated and stored in approved, regulated areas at the Site prior to shipment to ORNL.

Low Level, Transported to Oak Ridge National Laboratory

KAPL solid radioactive waste shipments to ORNL are subject to ICC regulations covering "transportation of explosives and other dangerous articles by land and water". Regulations pertinent to radioactive shipments are found in Sections 73.391 through 73.396 of Tariff No. 10, Agent H. A. Campbell's regulations, published by the Bureau of Explosives, Association of American Railroads. This shipment specifications agency was authorized by Act of Congress (62 Stat.738; 18 USCA 831-835) June 25, 1948.

Section 73.392, Paragraph "C", is specifically applied to KAPL radioactive waste shipment:

"Radioactive materials such as ores, residues, etc., of low activity packed in strong, tight containers are exempt from specification packaging and labeling requirements for shipment in carload lots by rail freight only provided the gammaradiation or equivalent levels will not exceed 10 milliroentgens per hour at a distance of 12 feet from any surfact of the car and that gamma radiation or equivalent will not exceed 10 milliroentgens per hour at a distance of 5 feet from either end surfact of the car..."



REFER TO KEYS ON FOLLOWING PAGES FOR EXPLANATION OF BUILDING NUMBERS AND NUMBERED MONITORING STATIONS

ENVIRONMENTAL MONITORING STATIONS AT KNOLLS SITE

Figure 2 (See Key on Following Page)

KAPL-1831

KEY TO FIGURE 2 - KNOLLS SITE ENVIRONMENTAL MONITORING LOCATIONS

(1) Continuously operating air monitor for assaying filterable air-borne radiocontaminants.

(2) Continuous air monitor with recording facilities for assaying filterable and gaseous air-borne radiocontaminants.

(3) Continuous recording, background radiation monitor.

(4) Integrating film and ionization chamber radiation monitors.

(5) Precipitation collection station with rain gauge and recorder.

(6) Vegetation and soil sampling station.

(7) Test wells around perimeter of underground high level radioactive waste storage tank farm.

(8) Sampling station for ground water from radioactive liquid waste processing building foundation and from underground high level radioactive waste storage tank farm.

(9) Shugg well, an abandoned private well at the south boundary of the Knolls Site.

(10) Sampling station for continuous and representative monitoring for radioactivity and chemical content in the sewer effluent from the Knolls Site to the Mohawk River.

(11) Meteorology station for continuously recording wind direction and velocity, lapse rate, relative humidity, and temperature.

(12) Standby iodine-131 and tritium air monitors.

"The term 'low activity material' is not defined by the ICC. The wording of 73.392 (c) (and 74.532(j)(i)) implies that any gamma emitting material, a full carload of which does not produce radiation in excess of 10 mr/hr at a distance of 12 feet from any surface of the car may be considered low activity for this purpose. Limitation of the radiation from either end of car to 10 mr per hour at 5 feet may be achieved by either spacing or by the use of shielding."[48]

Although KAPL waste is shipped under the above "exemption" clause with radiation levels held to within prescribed levels, waste is packaged in specification containers or containers for which KAPL holds a Bureau of Explosives special permit. Although the regulations allow exemption from specification packaging, this exemption is not claimed. However, utilization is made of the technical and practical experience of the Bureau of Explosives to assure incident free, safe transit of KAPL waste shipments. Wastes collections under Box Numbers 8 and 10 (Figure 1) are periodically transported via box cars to ORNL.

LIQUID WASTES

High Level, Underground Storage Tanks

An underground, reinforced concrete encased tank farm contains six stainless steel units (Figure 3). The wastes collected and stored in these tanks were generated primarily by fuel separations chemistry pilot plant operation at KAPL, which have since been discontinued. After considerable decay, about half of these wastes have been fed to the

(12) Standby Iodine-131 and effluent air monitors.

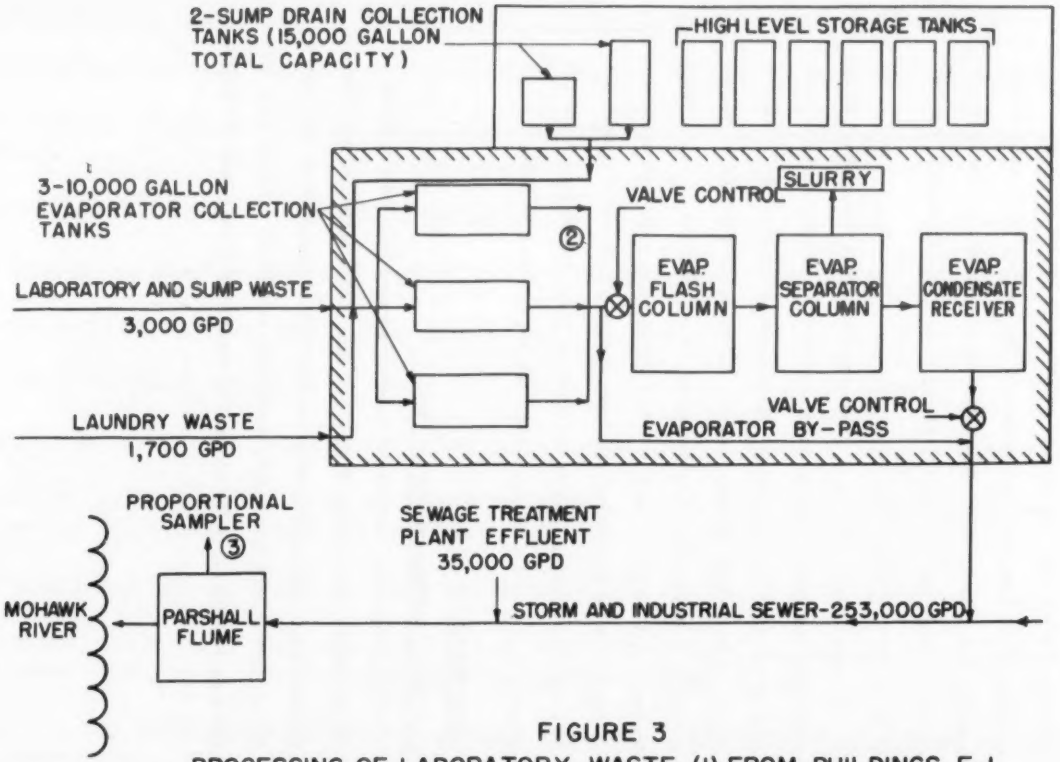


FIGURE 3

PROCESSING OF LABORATORY WASTE. (1) FROM BUILDINGS E-1, E-2, E-3, E-4, G-1, G-2, AND H3. (2) ANALYZED BEFORE RELEASE AT TWO LOCATIONS. (3) CONTINUOUS DAILY PROPORTIONAL SAMPLE ANALYZED.

evaporation system (Box No. 15, Figure 1) and shipped to ORNL. The remaining half is currently being processed for final disposal.

There is also an underground storage facility consisting of one, 55-gallon stainless steel drum located next to Building E-2. This drum receives liquid waste from dissolution work performed in a chemistry cell located in RML. This tank is placed in a concrete cask for on-Site handling and transportation. The waste has three possible paths (all box references are to Figure 1):

(a) Sampling of drum and dumping into Retention Tanks (Box 18),

(b) Sampling and dumping into high level storage tanks (Box 14), and

(c) Storage (Box 8) prior to shipment to ORNL in concrete cask.

High Level, Evaporation System

The evaporators (see Figure 2) are fed from two sets of tanks, one being the six stainless steel underground storage tanks (Box 14) and the other the three retention tanks (Box 18). The slurry is drummed (Box 16) and the condensate is discharged to the sewer after storage and monitoring.

High Level, Slurry Drumming

When the total solids in evaporator slurry reaches 80%, the hot slurry is remotely drawn off into 55-gallon, polyethylene-lined, carbon steel drums. The drummed slurry solidifies when cooled to room temperature. When a slurry drum weighs 500 pounds, it is moved to storage before shipment to ORNL. The radiation level on the outside of slurry drums is controlled by the use of an inner liner to contain the slurry and earth shielding between the liner and the outer drum.

RELEASE AT TWO LOCATIONS
PROPORTIONAL SAMPLE ANALYZED.

Low Level, Receiving Tanks

The retention tank system consists of three, 10,000-gallon, stainless steel hold tanks and a sump drain tank farm made up of one tank of 5,000 gallons and one of 10,000 gallons capacity. The waste from laboratory sinks and sumps is held up in the retention tank system and sampled when a tank is full. The radioactivity content of liquid waste from laboratory sinks and sumps normally permits release to the Mohawk River via the monitored sewer. The evaporator system (Box 15) is available if it becomes necessary to reduce radioactivity levels in this waste.

Lower Level, Release to Sewer System and Surface Streams

The flow of combined Site sewage discharge (industrial, storm, and sanitary) is continuously measured by a Parshall Flume and float chamber and proportionally sampled by a float regulated, two way solenoid valve in a steady-flow sample line (See Figure 4). The sewage flow is continuously recorded on seven day charts.

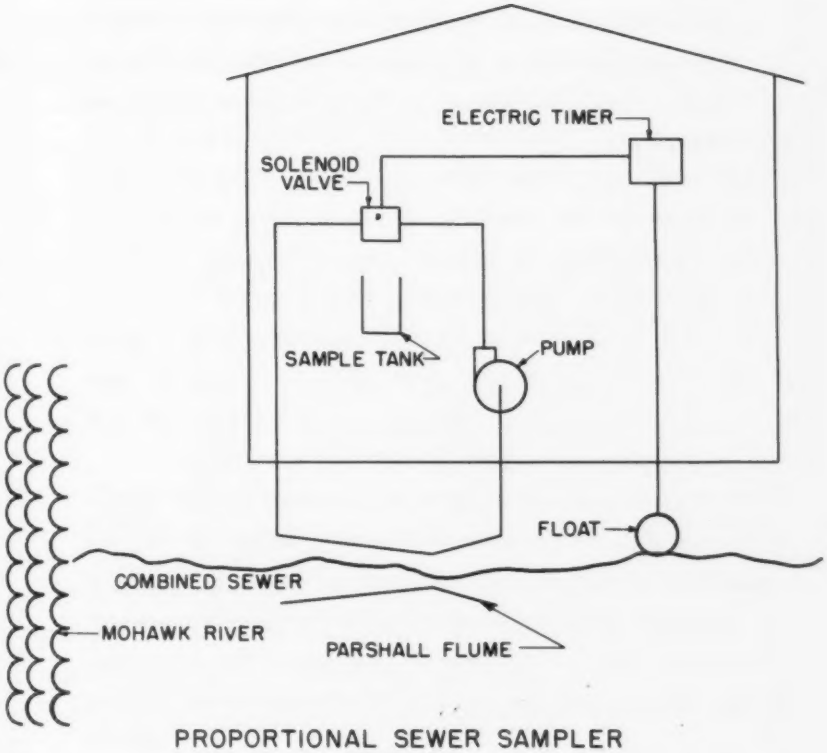
AIR-BORNE WASTES

Gases, Scrubbers

In cases where Radioactive Materials Laboratory operations produce volatile iodine and ruthenium fission products, caustic scrubbers and cold traps are incorporated in the venting system of the experimental or operating equipment.

Collection Tanks

When RML operations produce noble radioactive gases, a gas receiver is available to hold up gases until radiation level is determined. No hold up is necessary if dose-rate on gas receiver is less than 10 mr per hour.



PROPORTIONAL SEWER SAMPLER
FIG. 4

Particulate Filtration

Vent systems from laboratories handling radioactive materials contain roughing and high-efficiency filters which remove particles of sub micron size. The Special Nuclear Materials Machine Shop exhaust air passes through a cyclone separator and roughing filters to remove particulates before the air is released to the atmosphere. The waste filters from these systems are generally treated as low-level solid waste, Box 10.

FACILITIES AND PROCEDURES - WEST MILTON

Radioactive wastes are produced at the West Milton Site by the test operations of the S3G (Triton) prototype. A simplified flow diagram (see Figure 5) indicates the primary sources of solid, liquid and gaseous waste, the specific components and facilities involved in the processing of wastes, and the disposal of these wastes.

Solid Wastes

There is no provision for the final disposal of radioactive solid waste at the West Milton Site. Potential sources of low level waste are activities associated with the maintenance and operation of the reactor, and reactor power plant systems. The spent resin from primary coolant demineralizers constitutes the high level solid waste requiring shielded containers for handling and shipment. Fuel elements when spent, are packaged in shielded transport containers for shipment to an off-Site fuel reclamation plant.

The most probable radioactive contaminants of solid waste are activated corrosion products and other activated elements in the reactor primary coolant. These radioelements will be mainly accumulated in the primary coolant demineralizers. When removing or regenerating the demineralizers, the waste resin container will be removed and stored in a shielded area until radioactivity has decayed to a radiation level permitting more efficient handling, packaging, and shipment to ORNL.

Low level solid waste, either compressible or non-compressible, generated in contaminated areas is placed in 20-gallon waste cans lined with waterproof paper. These cans are emptied into 4' x 4' x 4' wooden boxes, which are in turn transported to the Knolls Site for radioactive waste processing (baling and shipment).

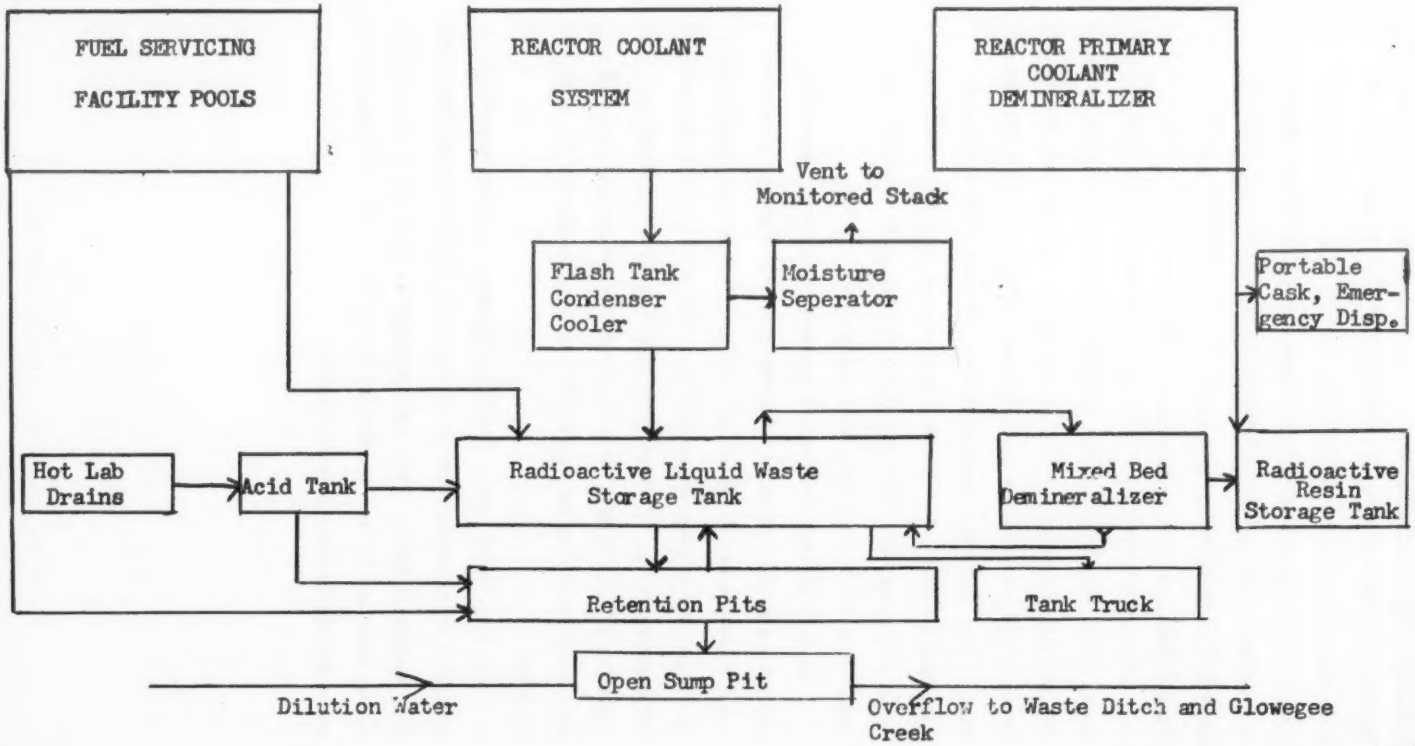


FIGURE 5 - RADIOACTIVE WASTE DIAGRAM OF KNOLLS ATOMIC POWER LABORATORY - WEST MILTON SITE

(Transportation over public highways complies with the USAEC, the Interstate Commerce Commission, and the Bureau of Explosives regulations.[18])

Liquid Wastes

There is no appreciable environmental dilution available for radioactive liquid waste in the small Glowegee Creek at the West Milton Site. Therefore, a Radioactive Liquid Waste Disposal System is provided to control and, if necessary reduce the radioactivity in liquid waste to permissible environmental limits before release to the Glowegee Creek.

Potential sources of radioactive liquid waste are Reactor Servicing Operations, which require the disposal of some of the water lost in the recirculating Reactor Primary Coolant System, and fuel servicing operations which involve the disassembly and packaging of fuel in the Fuel Service Facility water pool.

The radioactive liquid waste system provides a means to collect, sample, chemically treat, demineralize and dilute liquid waste for eventual release to the site stream whereever possible. When wastes warrant they are transported and disposed at the Knolls Site. The system includes a 5,000 gallon Radioactive Liquid Waste Storage Tank, a demineralizer of the mixed bed resin type, and a 1,000 gallon Radioactive Resin Storage Tank.

The Radioactive Liquid Waste Storage Tank receives Primary Coolant water when the normally recirculating Coolant System is interrupted for Reactor Servicing or when an overpressure in the Primary Coolant System is relieved through a pressure activated valve and drain system. Miscellaneous waste volumes may also be accumulated from the Fuel Service Facility water pool, the retention pits, the acid neutralizing pits, the radioactive demineralizer, the radioactive resin storage tank and

FIGURE 5 - RADIOACTIVE WASTE DIAGRAM OF KNOLLS ATOMIC POWER LABORATORY - WEST MILTON SITE

the moisture separator on the Primary Coolant flash tank vent. The retention pits collect all liquid waste to be released to the Clowegee Creek via the Site Ditch. If the radioactivity in retention pit water exceeds the operating environmental control limit of 2×10^{-6} μc/ml (refer to Page 773) and if plant water dilution is insufficient, the contents of retention pits are jetted to the Radioactive Liquid Waste Storage Tanks. The acid neutralizing pits collect all liquids from sinks and floor drains in laboratories and equipment decontamination rooms. If radioactivity in acid neutralizing pit liquids permit release to the Clowegee Creek, the contents are added to that of the retention pits. Otherwise, it is jetted to the Radioactive Liquid Waste Storage Tanks.

The contents of the Radioactive Liquid Waste Storage Tank may be sent to the retention pits for release to the Clowegee Creek, circulated through the demineralizer to reduce the concentration of certain radioactive elements, and, if demineralization is insufficient, drawn off into a tank truck and transported to the Knolls Site for disposal.

Air-Borne Wastes, Gases or Particulate Material

Radioactive gases or particulate material might be released from either the Fuel Service Facility or hull exhaust stacks. Absolute filters in the exhaust system of the Fuel Service Facility are utilized to reduce air-borne contaminants released to the environment. A vent to remove non-condensible gases from the Radioactive Liquid Waste Storage Tank, is valved to permit the controlled release of radio gases to the FSF stack at safe rates, as determined by the stack monitoring system.

COSTS

Capital equipment costs and operating and maintenance costs for the Knolls and West Milton Sites of the Knolls Atomic Power Laboratory are indicated on the following tables which are included in this section:

TABLE I - Capital costs of all facilities or systems involved in the collection, handling, processing, monitoring, and control of radioactive liquid, solid, and gaseous waste products.

TABLE II - Annual operating and maintenance costs for Knolls Site facilities, Fiscal Year 1958.

TABLE III - Operating and maintenance costs for radioactive waste management at the West Milton Site for a five-month period which is considered as representative of operations since the start-up of the S3G prototype plant.

TABLE I

CAPITAL COSTS OF ALL FACILITIES OR SYSTEMS INVOLVED IN THE COLLECTION, HANDLING, PROCESSING, MONITORING, AND CONTROL OF RADIOACTIVE LIQUID, SOLID, AND GASEOUS WASTE PRODUCTS

KNOLLS SITE	TYPE OF WASTE			TOTAL COSTS
	LIQUID	SOLID	GASEOUS	
*1. Bldg. H waste handling building	$ 324 000	$	$	$ 324 000
*2. Bldg. H evaporation plant & equip.	570 000			570 000
*3. Underground storage tanks	95 000			95 000
4. Retention tanks	29 539			29 539
5. Parshall flume	4 550			4 550
6. Sewer sampling instrumentation	4 450			4 450
7. Bldg. L-7, waste handling bldg.		19 300		19 300
8. Bldg. L-7, hydraulic baler		5 130		5 130
9. Area M-4, 250 hole underground storage pad including transfer crane		40 300		40 300
10. Bldg. K-6, intermediate level radioactive storage area		19 006		19 006
11. High level storage pit and cask storage area		11 668		11 668
*12. Bldg. G-3, vent scrubber house and equipment			40 000	40 000
**13. Bldg. D-3, cyclone separator and electrostatic precipitator			8 000	8 000
WEST MILTON SITE				
1. High level liquid waste facility	145 000			145 000
2. Low level liquid waste facility	47 000			47 000
TOTAL	$1 219 539	$95 404	$48 000	$1 362 943

* Originally designed for reactor fuel separations pilot plant.
** The electrostatic precipitator has been removed from service.

TABLE II

ANNUAL OPERATING AND MAINTENANCE COSTS FOR KNOLLS SITE FACILITIES
FISCAL YEAR 1958

	ITEM	TYPE OF WASTE			TOTAL COSTS
		LIQUID	SOLID	GASEOUS	
		$	$	$	$
1.	Labor inventory -- operational	8 686	8 686	4 346	21 718
2.	Overhead, 50% on operational labor	4 343	4 343	2 172	10 859
3.	Maintenance expense	7 778	1 100	600	9 478
4.	Tools and supplies	305	12 181	1 000	13 485
5.	Travel		182		182
6.	Telephone and telegraph		126		126
*7.	Health Physics charges, direct labor	4 350	1 270	3 500	9 120
*8.	Overhead, 50% on Health Physics direct labor	2 125	635	1 750	4 510
*9.	Chemistry analytical charges, including overhead	1 165			1 165
10.	Charges, shipment of wastes to ORNL		4 937		4 937
11.	Demurrage on railroad cars		62		62
12.	ORNL handling and storage charge		5 048		5 048
13.	License and fees	25			25
		$28 777	$38 570	$13 368	$80 715

* Includes environmental sampling.

TABLE III

OPERATING AND MAINTENANCE COSTS FOR RADIOACTIVE WASTE MANAGEMENT AT THE WEST MILTON SITE FOR A FIVE-MONTH PERIOD WHICH IS CONSIDERED AS REPRESENTATIVE OF OPERATIONS SINCE THE STARTUP OF THE S3G PROTOTYPE PLANT

	ITEM	TYPE OF WASTE			TOTAL COSTS
		LIQUID	SOLID	GASEOUS	
1.	Labor inventory, based on continuous operations, average man-week = 51.2 hours	$14 750	$ 95	$ 615	$15 460
2.	Overhead, 50% on operational labor	7 375	48	308	7 731
3.	Maintenance expense	750		100	850
4.	Tools and supplies	100	80	640	820
5.	Compressed gases	10			10
*6.	Health Physics--direct labor	464	232	1 416	2 112
*7.	Overhead, 50% on Health Physics direct labor	232	116	710	1 058
*8.	Chemistry analytical charges including overhead	333		30	363
9.	Transportation charges to KAPL	20	10		30
		$24 034	$ 581	$ 3 819	$28 434
	TOTALS EXTRAPOLATED ON AN ANNUAL BASIS (ESTIMATE)	$57 684	$1 392	$ 9 168	$68 244

* Includes environmental sampling.

ENVIRONMENTAL MONITORING

The Laboratory's liquid disposal and exhaust air systems, which release low levels of radioactivity to environmental surface waters and atmosphere, have been described in the section on "Waste Management Operations and Costs". It was pointed out then that all of this waste material is carefully monitored. This is done because of the need to:

1. Determine natural, fall-out and Laboratory operational influences on the environment.
2. Determine that operational controls on radioactive waste releases to the environment are adequate.
3. Determine that radioactivity levels in the environment satisfy the standards of the Atomic Energy Commission.

The Knolls and West Milton Site environmental monitoring programs were initially instituted when the Sites were constructed to determine the background radioactivity of Site drainage systems and atmosphere. This resulted from natural causes and fall-out from weapons testing. The Mohawk River monitoring program was extended to include aquatic life (fish and plankton) as increased amounts of radioactive elements were released into the stream.[5-11]

Knolls Site Surface Water - Mohawk River

The Mohawk River monitoring program includes sampling of the river water, river mud, fish and plankton. This extensive program was started to determine the effectiveness of the Mohawk River in diluting the combined sewer effluent of the Laboratory. Water is sampled daily in the sluiceway of the New York State power station at the Vischers Ferry Dam (1.7 miles downstream of the Knolls Site), and at the raw river water intake of the Cohoes Municipal water supply system. Samples from Cohoes are composited weekly for analyses. River water, fish, mud, and plankton are sampled once a month with the exception of

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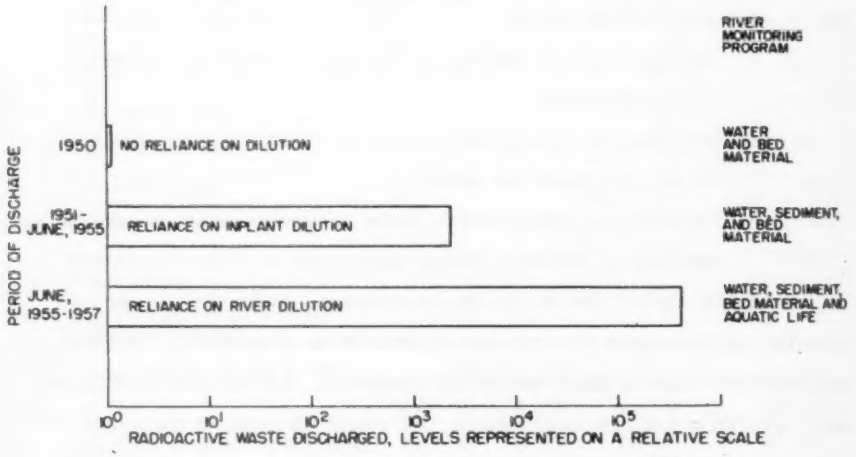
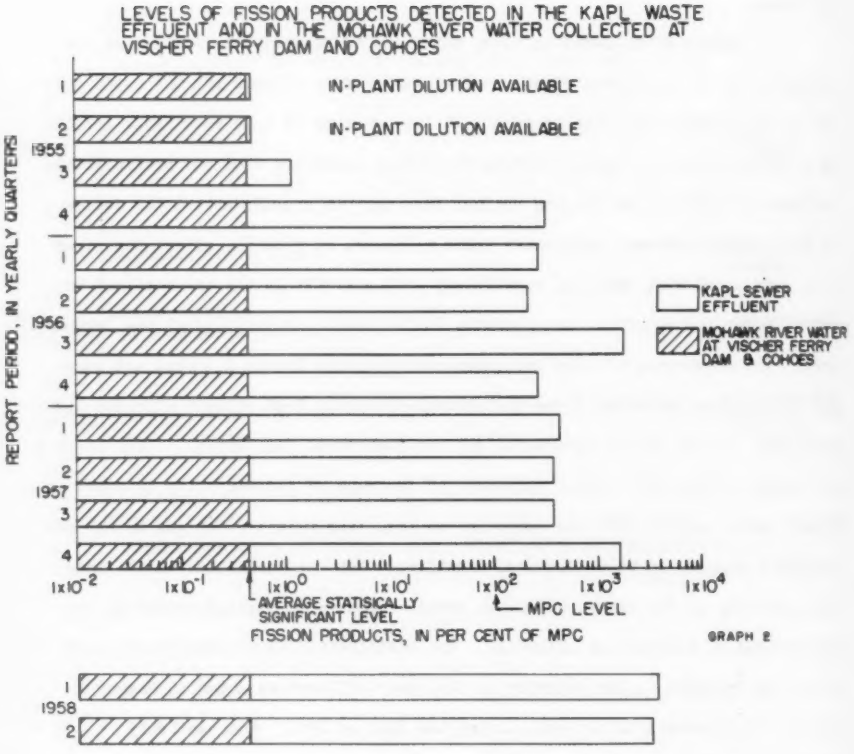
VARIATION OF RADIOACTIVE LIQUID WASTE DISCHARGED TO THE MOHAWK RIVER, DILUTION FACTORS, AND RIVER MONITORING PROGRAM

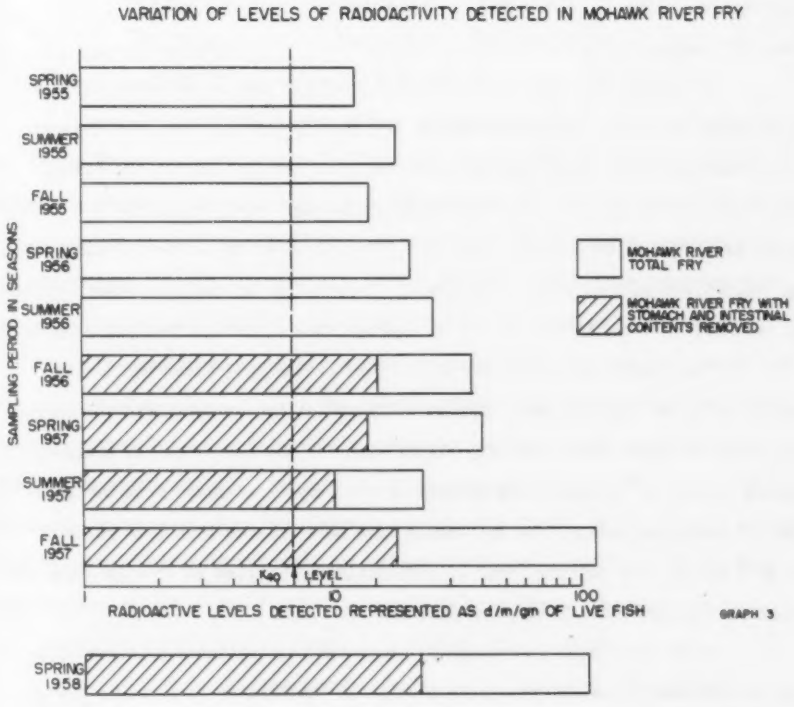


GRAPH I

the winter months. The fish are sampled at the Knolls Site sewer effluent where the activity levels are the highest. The other samples are obtained upstream and downstream of the sewer outlet. The samples of water, mud, and plankton are analyzed radiochemically for the determination of alpha and beta emitters. The fish are analyzed radiochemically for the determination of beta activity in tissue.

Effect of fall-out on river water and bed material is not large; the influence is greatest when Nevada nuclear tests are performed but the rapid decay of the fission products is such that the duration of this measurable effect is relatively short. Meteorological conditions involving wind and precipitation patterns in addition to varying nuclear test factors greatly affect the degree of fall-out influence. Laboratory waste releases, in general, affect the river in a very local way, that is, significant radioactivity levels are confined to water, river bed material, and plankton in the immediate vicinity of the Laboratory. The monitoring of fish indicates radioactivity levels although not harmful are in some instances above the natural-occurring K-40 content with the distribution varying in the structural and non-structural fish organs. (See Graph 3 and table on Page 28). Major increment of increase of radioactivity detected in Mohawk River aquatic life was noted in the fry, fish weighing 0.2 gms to 5.0 gms following changes in waste disposal practices. Fry were collected in the immediate vicinity of the sewer outfall to assure exposure to maximum potential concentrations of radioactive materials. The distribution of radioactivity in fry varies in relation to the activity in the sewer effluent as noted in Graphs 2 and 3. This is best illustrated during the Fall of 1957. Radioactivity in the structural tissues was constant but increased in the intestines and stomach.


LEVELS OF FISSION PRODUCTS DETECTED IN THE KAPL WASTE EFFLUENT AND IN THE MOHAWK RIVER WATER COLLECTED AT VISCHER FERRY DAM AND COHOES
REPORT PERIOD, IN YEARLY QUARTERS
1955
1956
1957
1958
1
2
3
4
IN-PLANT DILUTION AVAILABLE
IN-PLANT DILUTION AVAILABLE
KAPL SEWER EFFLUENT
MOHAWK RIVER WATER AT VISCHER FERRY DAM & COHOES
1x10-2
1x10-1
1x100
1x101
1x102
1x103
1x104
AVERAGE STATISICALLY SIGNIFICANT LEVEL
MPC LEVEL
FISSION PRODUCTS, IN PER CENT OF MPC
GRAPH 2


VARIATION OF LEVELS OF RADIOACTIVITY DETECTED IN MOHAWK RIVER FRY
SAMPLING PERIOD IN SEASONS
SPRING 1955
SUMMER 1955
FALL 1955
SPRING 1956
SUMMER 1956
FALL 1956
SPRING 1957
SUMMER 1957
FALL 1957
SPRING 1958
MOHAWK RIVER TOTAL FRY
MOHAWK RIVER FRY WITH STOMACH AND INTESTINAL CONTENTS REMOVED
Kaq 4 LEVEL
10
100
RADIOACTIVE LEVELS DETECTED REPRESENTED AS d/m/gm OF LIVE FISH
GRAPH 3

AVERAGE RELATIVE LEVELS OF RADIOACTIVITY DETECTED IN MOHAWK RIVER FISH

Period of Collection	River Fish Beta Radioactivity Levels, in $\mu c/gm$		
	Fry[a]	Small Specimens[b]	Large Specimens[c]
Prior to June, 1955	5.9×10^{-6}	2.7×10^{-6}	1.1×10^{-5}
June, 1955-June, 1958	1.5×10^{-5}	4.5×10^{-6}	1.8×10^{-5}

A typical year (1957) has been selected for the period since 1955, when utilization was made of the Mohawk River for dilution of low-level radioactive waste from the Knolls Site, to represent environmental findings in the River. In 1957, 14 curies of fission products (at an average concentration of 3.6×10^{-5} $\mu c/ml$) were discharged from KAPL in 1,410,000 gallons of Laboratory liquid waste and 102,300,000 gallons of sewer effluent. This volume of combined sewer effluent was released from the Knolls Site at an average flow of 200 gallons per minute.

The results of radiochemical analyses of the 1957 River monitoring program indicate that the water samples collected at Cohoes were less than the value that is considered physically significant for the radiochemical analyses applied (4.0×10^{-8} $\mu c/ml$). One percent of the samples analyzed from the location 1.7 miles downstream from the Knolls Site were greater than this value of 4×10^{-8} $\mu c/ml$. The maximum level of radioactivity detected at this location was 50% of the maximum permissible concentration for Sr-90.

River monitoring (see Graph 2) indicates that reliance on river dilution has affected water radioactivity levels at the Vischer Ferry Power Dam to

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[a]The term fry, as stated here, refers to specimens of fish varying in size from 0.2 gms to 5.0 gms of live-weight tissue. These specimens are collected as near the sewer outlet as possible.

[b]The term small fish, as stated here, refers to specimens of fish varying in size from 5.0 gms to 100. gms of live-weight tissue.

[c]The term large fish, as stated here, refers to specimens of fish greater than 100 gms in weight.

a very small degree and at Cohoes to a negligible degree . Results of analyses of Mohawk River bed material for the year, 1957, are presented below.

LEVELS OF RADIOACTIVITY DETECTED IN MOHAWK RIVER BED MATERIAL IN 1957

Sample Location	Radioactivity Levels, in d/m/gm. Alpha	Beta
Sewer Outfall (average)	1.70 ± 0.07	2700. ± 47.
(maximum)	17.0 ± 0.3	89,000. ± 1300.
500' Downstream (average)	0.28 ± 0.03	18.2 ± 1.5
(maximum)	0.83 ± 0.05	75.0 ± 5.0

As stated previously, the effect of the Knolls Site sewer effluent on the Mohawk River has been limited to the samples collected near the sewer outfall. It may be noted, above, that the samples collected 500 feet downstream from the sewer outfall depicted levels of radioactivity similar to samples collected upstream (18.2 ± 1.5 d/m/gm). This localized river bottom radioactivity doesn't affect appreciably the pick-up of activity by the fish in the river.

The results of plankton sampling collected in 1957 have been summarized and are presented below. The results of the alpha radioactivity analyses indicate normal levels as detected in specimens collected upstream from the Knolls Site sewer outfall. The result of maximum beta radioactivity analysis indicates an increase over the upstream samples by a factor of 15. This localized river plankton radioactivity does not affect the larger sized fish and has only a transitory affect on the fry in the immediate vicinity of the sewer outfall.

LEVELS OF RADIOACTIVITY DETECTED IN MOHAWK RIVER PLANKTON IN 1957

Sample Location	Radioactivity Levels, in 10^{-5} d/m/ml of River Water Alpha	Beta
Sewer outfall (average)	1.8 ± 1.3	78. ± 22.
(maximum)	3.1 ± 1.9	445. ± 51.
Downstream 2500' (average)	2.8 ± 1.6	32. ± 15.
(maximum)	4.5 ± 1.9	90. ± 32.

Assessment of the radioactivity levels in the Mohawk River and its aquatic life, relative to the criteria of the National Committee on Radiation Protection and Measurements, approved by the Atomic Energy Commission, and to the classification of the River by the New York State Water Pollution Control Board, discloses that releases of low level radioactive wastes to the Mohawk River do not adversely affect the river.

Knolls Site - Atmosphere

The exhaust air from Laboratory sources of potential environmental atmospheric contamination is continuously sampled to determine the radioactivity content. Weekly samples of stack air effluent are collected and analyzed for alpha and beta radioactivity content. Air samples are obtained by filtration through efficient filter papers. The samples of air effluent from the Radioactive Materials Laboratory is drawn through a caustic scrubber during operations which may release radioiodine to the atmosphere, and the liquid scrubber solution is radiochemically analyzed for radioiodine content.

Environmental air is sampled continuously at locations shown in Figure 2. Weekly filter samples are analyzed to determine the levels of particulate alpha and beta radioactivity in the air. Film dosimeters and a continuous Geiger Mueller beta-gamma detector also monitor radiation levels at these environmental monitoring stations. Air samples and monitors indicate the influences of natural and fall-out radioactivity as well as operational releases of radiogases. Comparison of results from air monitoring stations at both the Knolls and West Milton Sites is one means of determining the influence and pattern of natural and fall-out radioactivity levels.

TABLE 4 - AVERAGE ATMOSPHERIC MONITORING RESULTS

Year	Quarter	Environmental Air Alpha (10^{-14} μc/cc)	Environmental Air Beta (10^{-12} μc/cc)	Gamma Dose Rates Film Dosimetry (mr/hr)	Beta Gamma Radiation levels (GM counts/min)
1957	1	< 2	< 0.5	< 0.02	93
	2	< 5	< 1.1	< 0.02	105
	3	< 5	< 1.2	< 0.02	155
	4	< 6	< 0.7	< 0.04	132
1958	1	< 6	< 0.5	< 0.04	108
	2	< 6	< 1.8	< 0.03	139

The above data in Table 4 indicate that alpha and beta concentrations of environmental air are well below permissible limits from the inhalation consideration and the average general radiation levels for external irradiation are approximately at natural background levels and well within permissible limits. In fact, the frequent evidence of beta particulate activity in the atmosphere is associa- with fall-out rather than operational sources. Increases in general radiation level at the site are associated primarily with the transfer and storage of radioactive wastes rather than fall-out.

Environmental air and vegetation is sampled downwind from the RML ventilation exhaust stacks for radioiodine content during and following operations which are potential sources of contamination to the atmosphere. Only occasionally, in the case of iodine released from irradiated nuclear fuel examinations in the Radioactive Materials Laboratory, has there been reliance on atmospheric dilution to satisfy environmental radioactivity limits. The amount of atmospheric dilution available has been calculated by means of Suttons equations, and has been found adequate for normal operations. The amount of radioiodine

detected in environmental air and vegetation has never exceeded prescribed limits and presents no problem. The maximum I-131 concentration on Site in this period was 1.2×10^{-10} μc/cc of air and 2.3×10^{-5} μc/gm of vegetation. The I-131 vegetation average levels did not exceed permissible limits as measured by inhalation or ingestion intake on a continuing basis.

West Milton Site - Surface Water - Glowegee Creek

The Glowegee Creek monitoring program is not as extensive as the Mohawk River program. This is because the creek has no reliable dilution potential for Site liquid waste, and levels of radioactivity in waste released to it are operationally controlled to maximum permissible environmental levels prior to discharge.

Water samples are taken weekly of liquid effluent in the West Milton Site ditch and of creek water above and below the point where the Site ditch and the creek waters come together. The samples are analyzed for alpha and beta activity levels. The following table presents the average radioactivity content found in samples taken during operation of the S3G Facility from August to December, 1958.

	Creek Above Site Ditch Outlet	Site Ditch at Outlet to Creek	Creek Below Site Ditch Outlet
Alpha (Avg)	2.7×10^{-9} μc/ml	2.7×10^{-9} μc/ml	2.7×10^{-9} μc/ml
Beta (Avg)	2.5×10^{-8} μc/ml	4.3×10^{-8} μc/ml	4.3×10^{-8} μc/ml
(Max)	(7.3×10^{-8} μc/ml)*	(1.1×10^{-7} μc/ml)	(8.6×10^{-8} μc/ml)

Site Operations have not influenced the levels of alpha activity in the Site ditch or Glowegee Creek. The above-average levels of beta activity found upstream are attributed to radioactivity from fallout*deposited in the stream during and as a result of precipitation and are not due to operational

*Beta activity concentrations in precipitation samples obtained during periods of increased Creek radioactivity ranged up to a maximum of 4×10^{-6} μc/ml.

influence. Maximum concentrations determined in the Site ditch at the outlet to the Glowegee Creek were well below the operational limit 2×10^{-6} (see Page 6).

West Milton Site - Atmosphere

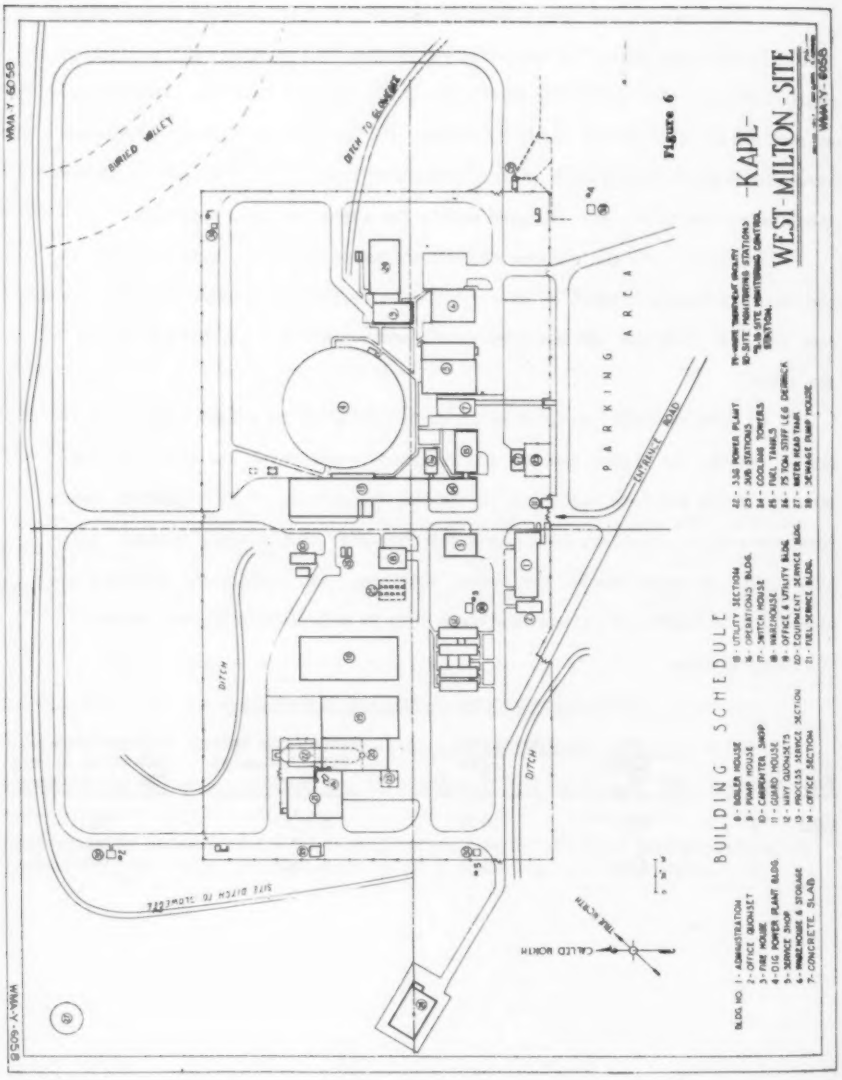
The exhaust air from potential sources of environmental contamination is continuously monitored to determine particulate and gaseous beta radioactivity concentrations. The air contamination monitors can shut down the ventilation exhaust system at a preset level to prevent the exposure of persons in the environment when the level goes above recommended limits. Filter samples of particulate radioactivity are analyzed weekly for alpha and beta activity.

Particulate and gaseous concentrations of alpha and beta activity in the vapors released present no on- or off-site radiation exposure problem, since they are less than the maximum permissible environmental level before leaving the stack.

Environmental air is sampled continuously at locations shown in Figure 6. Weekly air filter samples are analyzed to determine the levels of particulate alpha and beta radioactivity in the atmosphere. Film dosimeters and a continuous Geiger Mueller, beta gamma detector also continuously monitor radiation levels at environmental monitoring stations. Air samples and monitors indicate the influence of natural and fall out, as well as operational releases of radioactivity.

Average Environmental Monitoring Results

Year	Quarter	Air Concentrations Alpha (10^{-11} μc/cc)	Air Concentrations Beta (10^{-12} μc/cc)	Gamma Dose Rates Film dosimetry (mr/hr)	Beta Gamma Radiation levels (GM counts/min)
1958	3	< 6	0.5	< 0.02	56
	4	< 6	1.0	< 0.02	59


WAA-Y-6059
WAA-Y-6058
-KAPL-
WEST-MILTON-SITE
Figure 6
BUILDING SCHEDULE
PARKING AREA
DUFFIELD ROAD
DITCH
DITCH TO SLOWLEY
BURIED VALLEY
CALLED NORTH
TRUE NORTH

Every case of a statistically significant indication of beta activity can be attributed primarily to fall out. Neither the air concentrations nor the general site radiation levels exceed environmental standards recommended by the National Committee on Radiation Protection.

RESEARCH AND DEVELOPMENT

Research and Development activities in the waste management area have benn directed toward increasing the knowledge of the KAPL physical environment, and the development of technical methods to assess environmental hazards occasioned by the release of radioelements. Major studies in these areas have been performed by the U. S. Geological Survey and KAPL Health Physics.

The U. S. Geological Survey has studied the hydraulic and hydrological characteristics of the Mohawk River in the reaches from Schenectady, N. Y., to Cohoes, N. Y. These river studies encompassed flow patterns, water temperature, sediment loading and characteristics, chemical quality of river water, bed material; and tracer diffusion patterns, including radioactivity tracing with P-32. The latter investigation was made in 1953-1954, and representatives from the New York State Health Department, Harvard University, and KAPL participated actively in it. The results of these studies will be reported in a U. S. Geological Survey report titled Hydrology and Tracer Studies of the Mohawk River at the Knolls Atomic Power Laboratory.

These studies aided materially in the decision to utilize the Mohawk River for dilution and final disposal of large volumes of low level radioactive liquid wastes from the Laboratory. They showed that such disposition could be made in complete conformance with established national and state standards.

The Health Physics studies were directed primarily to obtain technical information required to assure the adequacy of methods employed to evaluate and assess radioactivity hazards in the environment. Investigations were made to

determine two main factors. The first was the K-40 content, and Strontium-Calcium ratios in the Mohawk River water and fish, as they effect the interpretations of the Sr-89 and Sr-90 hazard in drinking water or in consuming fish. The second was the efficiency of filter media used for monitoring the atmosphere for concentration of particulate radioactivity.[47] Collection efficiencies were investigated for particle sizes in the electron microscope range.

The U. S. Geological Survey geological studies at KAPL were directed to the selections of a favorable site for construction of the concrete underground cells containing tanks for the storage of high level radioactive waste.

The U. S. Geological Survey programs and assistance are used to provide the technical information which aids in assuring safe and economical radioactive waste disposal to the environment.

January 13, 1959

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RADIOACTIVE WASTE DISPOSAL PRACTICES AT LOS ALAMOS

(By C. W. Christenson [1])

The operation of any facility processing radioactive materials necessarily produces an unwelcome waste material. The solid wastes consist of such things as protective gloves, laboratory equipment, and almost anything else used in the processing. Liquid wastes are derived from water used in various steps of processing radioactive isotopes, laboratory wastes, cleansing operations, etc. The airborne wastes arise from off gases from various processes as well as the air exhausted from dry boxes and hoods.

The disposal of radioactive wastes is different from all other waste disposal practices because of the very nature of the materials involved. Unlike other waste materials which may be changed or destroyed, the radioactive material is affected only by time, and with those of great concern, the time required for decay is too long. The only two methods available are concentration and containment or dilution to a point of minimum hazard. Obviously, concentration and containment is the method of choice because, if properly accomplished, this does not increase the total amount of radiation in the environment. There are, however, circumstances in which it is more feasible to combine the two methods. At Los Alamos nearly all of the waste products are concentrated and confined to well-regulated burial pits.

The hazard from radioactive wastes arises from the radiation involved which is contributed by alpha, beta, and gamma rays or particles. The three types of radiation are extremely toxic within the body but they vary in penetrating power and the degree of external radiation hazard they create. The alpha particle is quite large and its penetrating ability is, therefore, very small—the particle will not penetrate a sheet of paper. The beta particle is more penetrating, and if one is dealing with large sources of gamma-emitting isotopes several feet of concrete shielding is necessary for protection of operating personnel.

The waste-disposal problem at Los Alamos is somewhat different from that at most of the AEC installations because the major radioactive isotope involved is plutonium. Plutonium is, perhaps, the most toxic radioactive element known to man and extreme care must be used when dealing with this isotope. It is, however, an alpha emitter and thus requires no shielding or remote control operation as is the case with gamma-emitting isotopes.

The waste-disposal philosophy at Los Alamos has always been to discharge the least amount of radioactivity to the environment possible with available equipment and to meet or exceed current medical standards for radioactivity in air or water. Improvements have been made continually since the early, hectic, days of production of the first bomb. The areas in and around Los Alamos have been monitored continually and there has been no noticeable deleterious effect on the environment.

The methods of control and containment of the wastes will be taken up in the order of solids, liquids, and airborne wastes.

SOLID WASTES

It must be assumed that the equipment used in processing radioactive material sooner or later will become contaminated to the point at which it is a health hazard to operating personnel. Contamination is not only a health hazard but may represent a possible source of error in analytical procedures. Every effort is made to clean or decontaminate this equipment so that it may be reused. The decontamination section of the monitoring group cleans everything from

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Control chemist, Hospital Liquids, Inc., Chicago, 1939–40.
Chief chemist, sewage treatment plant, Dallas, Tex., 1940–41.
Superintendent of sewage plant and chief chemist, water department, El Paso, Tex., 1941–42.
First lieutenant to major, Sanitary Corps, Army of the United States, 1942–46.
Superintendent of water and sewage treatment plants, El Paso, Tex., 1946–48.
Assistant leader, industrial waste treatment group, Los Alamos, N. Mex., 1948–53.
Leader of same group, 1953 to date. Present title: Group leader, H–7, Health Division, Los Alamos Scientific Laboratory.
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platinum disks to five-ton trucks. However, certain materials cannot be decontaminated economically and these must necessarily be disposed of by burial or other means.

The disposal of contaminated solid wastes has followed essentially the same pattern since the inception of the project. There have, of course, been refinements in techniques to eliminate hazards. Small items such as laboratory glassware, contaminated rags, paper, and rubber gloves, small machine parts, etc., are collected in plastic bags and placed in cardboard cartons, sealed with masking tape and hauled by truck to a protected dump used only for disposal of contaminated material. Each box is marked as to date and area from which it was received. Larger pieces of equipment are generally sealed in plastic or painted to contain any contamination and hauled in that condition. In one case, an entire building which could not be satisfactorily decontaminated, D building, where the first visible piece of plutonium was produced, was dismantled, carefully transported to the dump, and buried.

The dumps are large pits about 500 feet long, 100 feet wide, and 25 feet deep excavated in remote areas. The locations are chosen with the help of U.S. Geological Survey personnel who determine the geological features of the area such as geologic faults, drainage, etc., in order to find the most suitable site for the best containment of radioactive material. The trash is placed in the dump and covered with about one-half foot of dirt after a 2-foot layer of trash has accumulated. When the layers approach the top a 4-foot layer of dirt is placed on the surface. Figure 1 shows the method of burial.

The pits are accurately surveyed and the location of each daily accumulation is recorded as to its exact position in the pit. The various pits have been located and mapped accurately. Concrete posts permanently identify the sites. Each pit is fenced with security-type fence. The gates are kept locked and only authorized personnel are permitted to enter the area.

From 1943 to date, some 35 acres of land have been utilized for six different pits and a 40-acre plot has recently been set aside for future use. A picture of the pit now being used is shown in figure 2. The material on the left has been placed in the pit and will be covered after the 2-foot layer is completed. The materials in the center of the picture are slightly contaminated parts of an obsolete building which will be burned before burial. The cost of monitoring the lumber exceeds the value of the material so it is deemed more economical to burn it to save burial space.

One of the sites at Los Alamos uses rather large amounts of radioactive barium and lanthanum (Rala). As the half life of this material is quite short (12 8 days) the sources soon become too dilute for the experimental work. These sources also contain from about 0.1 to 5 curies of strontium 89-strontium 90. When the activity has decayed to less than 20 curies of the Rala material it is disposed of at the contaminated dump. The material is placed in a Dural container which is in turn placed in a 4-inch diameter steel casing. The steel casing is then placed in a lead pig for radiation protection before transfer to the dump. Several holes 2 feet in diameter and 12 feet deep have been drilled inside the fenced area of the contaminated sump. A tripod and boom arrangement is used to move the container from the lead pig to the drilled hole. Dirt is placed in the hole after burial so as to maintain the radiation at less than 1 mr./hr. at the surface. This operation has been performed several times with none of the operators receiving more than about 10 mr. radiation dose.

The annual operating costs for solid waste disposal for the year of 1957 are as follows:

Materials

	Cost
Boxes, barrels, plastic bags, etc	$9,000
Monitoring	8,000
Escort vehicle	450
Janitorial service in filling boxes, bags, etc	29,000
Pickup service, truck to dump	15,000
Cover trash at dump	5,500
Excavation	7,200
Fencing	400
Total	74,000
Total volume buried (cubic yards)	3,187
Cost per cubic yard	$23.50

Figure 1

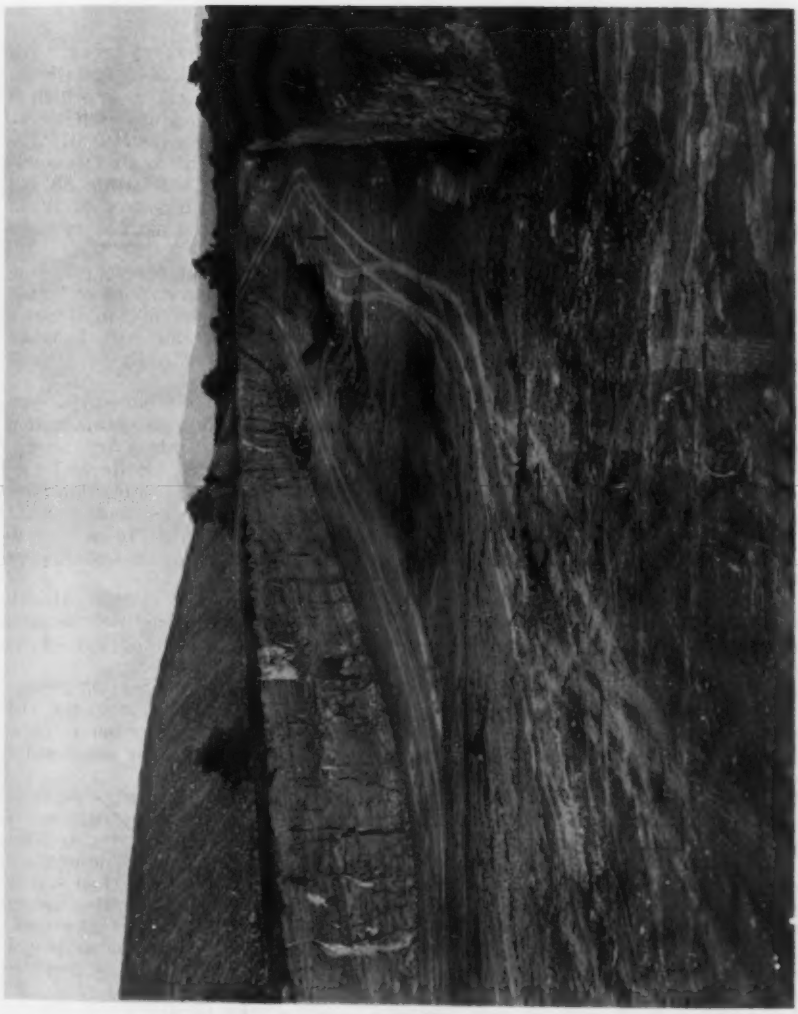


Figure 2

Figure 1

It is felt that the operation of the burial pits at Los Alamos has been accomplished with a minimum health hazard. The radioactive material placed in the pits is generally quite insoluble or rather well fixed to the surface of solid materials, hence it will be contained quite well. It has been demonstrated that plutonium in a soluble form, as well as other isotopes used at Los Alamos, is very well retained in the local tuff and it can be expected that the radioactivity will be trapped in the tuff before it could reach the underground water supplies. Furthermore, the Los Alamos average of 18 inches of rainfall per year provides very little water for transport of radioactivity to lower depths.

LIQUID WASTES

Liquid wastes at Los Alamos are contributed from four principal areas. The major contribution comes from the various chemical, physical, and metallurgical laboratories on the project. Another source is the DP west area which is concerned with the formation, fabrication, and recovery of plutonium metal. In these first two areas plutonium is the major radioactive constituent. The third area is Ten Site where the work is largely concerned with high concentrations of radioactive barium-lanthanum mixtures with some strontium 89 and small amounts of strontium 90 as contaminants. The fourth source of liquid wastes is the contaminated clothing laundry which is concerned with all of these types of contamination.

The allowable concentration of plutonium in drinking water for general consumption is in the order of 1 ounce of plutonium in 100 million gallons of water. This quantity of water is approximately the average daily consumption of water for Washington, D.C. Plutonium is very valuable and extreme care is taken to avoid losses, so that the total amount to be disposed of is actually very small and dissolved or suspended in a large volume of water.

Laboratory wastes.—The wastes contributed by the various laboratories consist of waste solutions, wash water, and waste water from decontamination procedures. Prior to 1950 all of these wastes were discharged into a dry canyon. This canyon was fenced throughout its length on the project. Soils and subsurface water (when available) have been monitored continually since that time to ascertain the location and level of contamination. There has been a slight movement of activity down the canyon but the level is not believed to be hazardous and all of the activity is contained within the confines of the Los Alamos project.

A group of engineers and chemists, together with the U.S. Public Health Service, started research on methods of treating liquid wastes in 1948; by 1950 a treatment plant was completed and put into operation. A photograph of the plant operating room is attached as figure 3.

The plant was built at a cost of $250,000 and was designed to treat 35 gallons per minute. Since that time the waste flows have increased markedly and it has been necessary to increase treatment capacity to 120 gallons per minute. It is expected that the plant will be able to handle waste flows for some years to come.

The method of waste treatment for removal of plutonium is much the same as that used in conventional water treatment plants except that the treatment is somewhat more rigorous than is generally employed in municipal systems. This treatment consists of adding an iron salt (ferric sulfate) and lime. The mixture of iron salt and lime forms a precipitate (ferric hydroxide) which then settles out to the bottom of the settling tank. The precipitate (sludge) carries nearly all of the plutonium with it and it is removed from the bottom of the tank, dewatered and placed in 55-gallon drums before being taken to the contaminated dump for burial. It has been determined that the plutonium, in this form, is very resistant to leaching by water.

Chemical additions must be very carefully adjusted and closely controlled because the composition of the wastes frequently varies over a wide range. The precipitation of the ferric hydroxide is sometimes quite difficult because of the presence in the waste of large amounts of soaps and detergents which have been used for cleanup operations. Excessive amounts of cleaning reagents sometimes require additional chemicals and refinements in treatment.

The above wastes, when received at the plant, contain about 20 times the maximum permissible concentration of plutonium in drinking water. After treatment the water is monitored before discharge to the canyon and if the radioactivity is above acceptable levels it is recirculated and treated again. Over the years the average plutonium content of the discharged treated wastes has been something less than 10 percent of the allowable concentration.

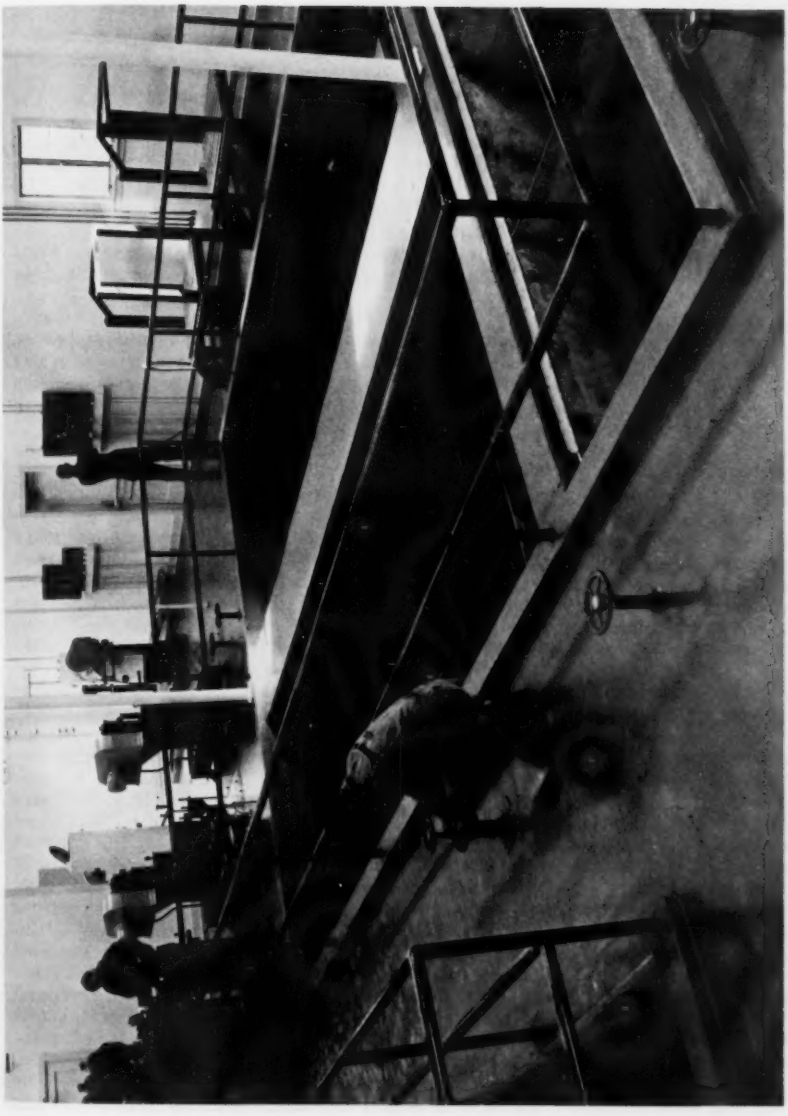


Figure 3

DP west wastes.—These wastes result from process water used in formation, fabrication and recovery of rather large amounts of plutonium metal.

Although the volume of wastes produced at this side is considerably less than that from other areas, the concentration of plutonium is much higher—about 100 times the allowable drinking water tolerance. The concentration of inert constituents is likewise much higher and treatment of the waste is much more critical.

Prior to 1952 these wastes were discharged into a seepage pit some 20 feet deep. During this period the pit was able to absorb all of the wastes and repeated monitoring since that time has shown that the radioactivity has not moved from the site of discharge. By 1951 the volume of wastes increased to a point where the pit was not able to absorb all of the wastes. Accordingly, a plant very similar to the one described above was designed and put into operation.

The method of treatment is essentially the same as that used for treatment of laboratory wastes except that even more care in operation is required. The average discharge from this plant during the 5 years of operation has been something less than one-tenth of the allowable concentration of plutonium in drinking water.

In addition to the main waste stream this area produces a variety of wastes which are necessarily treated separately because of the high concentration of inert ions which would interfere with treatment of the main waste stream. Some of these wastes are derived from recovery operations and contain an extremely high concentration of inert salts such as iron and aluminum as well as plutonium. The neutralization of hydrofluoric acid used in processing plutonium yields yet another waste. These wastes are treated by neutralization with acid or alkali and the addition of calcium chloride. Here again, precipitates are formed which carry the plutonium down with them. The resulting sludge is then dewatered and the sludge cake is placed in drums and transported to the contaminated dump for burial.

Ten site.—The liquid wastes produced at this site result from process water and cleanup operations in the "hot" cell used for processing large sources of radioactive barium-lanthanum. This waste contains rather high concentrations of short-lived beta- and gamma-emitting isotopes and some strontium 90 as impurities. The original plan of treatment was to provide storage to allow for decay of the short-lived nuclides before discharge to a canyon. Four 50,000-gallon tanks were installed, one of which was utilized for the air cleaning process to be described later. These tanks served admirably for a few years but it became apparent that expanded operations and increased waste flow would exceed the capacity of the tank farm.

Accordingly, a research program was initiated to determine the best method of treating these wastes. The result of this program was the installation of an ion exchange plant in 1955. The principle of treatment is much the same as that used in home water softeners. The water flows over a synthetic ion exchange material and metallic ions (whether radioactive or not) such as: calcium, iron, barium, lanthanum, and strontium are transferred and bound to the ion exchange material. After several thousand gallons have passed through the columns the ion exchange material becomes exhausted and must be regenerated. In home water softeners a solution of common salt is used but salt is not satisfactory for this installation and a more stringent regenerant must be used—nitric acid. The acid removes the nuclides as well as other ions from the ion exchange material so that the ion exchange cycle can be repeated. The radioactivity is removed from the spent regenerant by treatment with lime, iron, and soda ash. The mixture of these chemicals form a precipitate which settles to the bottom of the treatment tank carrying with it nearly all of the radionuclides. The resultant sludge is then mixed with about twice its weight of cement in a 55-gallon drum and taken to the contaminated dump for burial. The sludge-cement mixture has been found to be very resistant to leaching by water.

As can be seen in figure 4 the plant is considerably more complicated than is a home water softener. Lead bricks are placed to surround the different units in the plant to shield the operators from gamma radiation produced by the barium-lanthanum mixture. The plant is actually two ion exchange units operated in series to provide more efficient removal of radioactive ions. The other equipment in the picture is used to provide preliminary chemical treatment and control of various factors necessary for efficient treatment.

Many changes have been made in the plant during the past 3 years to improve efficiency of operation and to provide for greater treatment capacity. The results of treatment have been satisfactory and, taking available dilution into consideration, the discharges are well below the allowable concentration in drinking water.

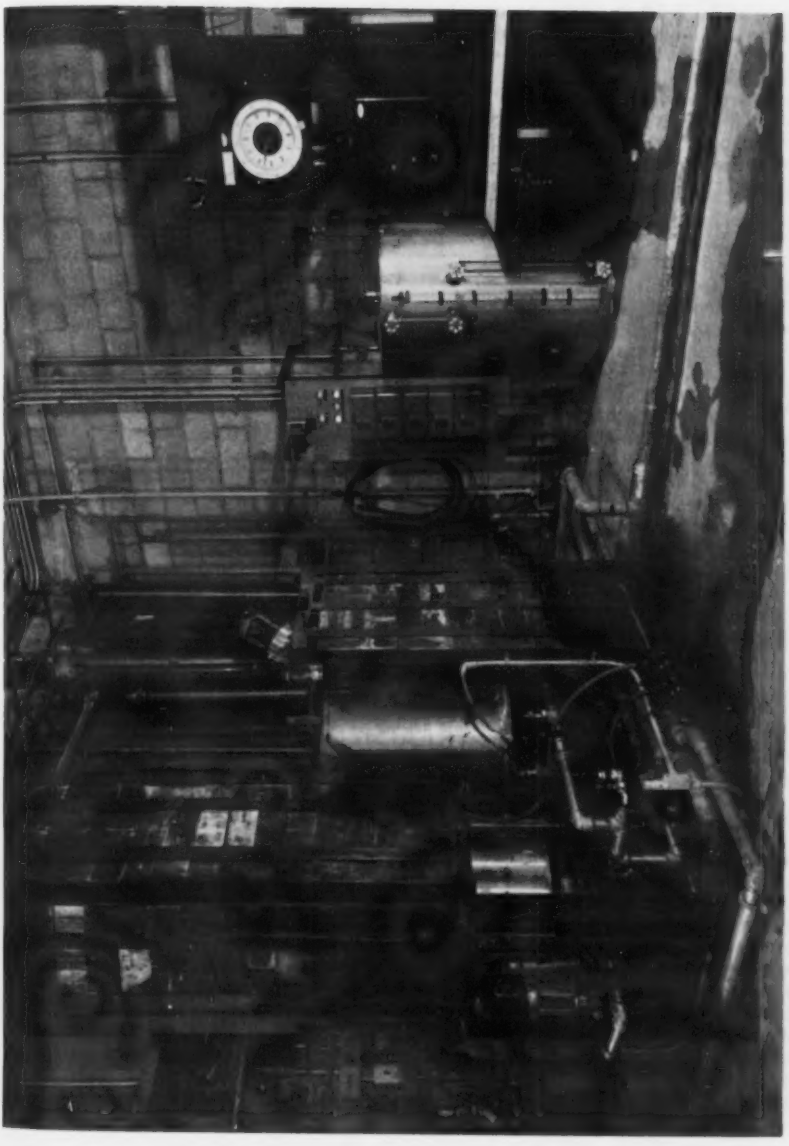


Figure 4

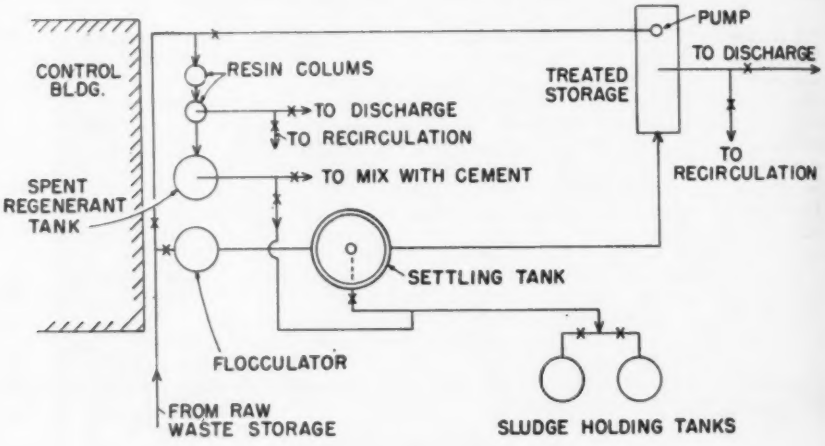
In anticipation of additional site operations and further increased waste flow, a new plant has been designed to replace the existing unit. It will include chemical pretreatment and much higher capacity ion exchange columns. All of the treatment tanks, sludge holding tanks, and ion exchange units will be buried underground to minify radiation exposure of the operation.

A flow diagram of the plant is shown in figure 5. The chemical treatment is much the same as was described under TA–45 and DP West wastes. Here the sludge will be held in underground tanks for several months before filtration and burial. The effluent from the chemical pretreatment will be pumped through the ion exchange column before discharge to the adjacent canyon.

It is important to have the chemical pretreatment since there is a possibility that some plutonium might find its way into the waste, and plutonium is not removed efficiently by ion exchange treatment.

It is also possible to bypass the chemical pretreatment and treat the raw wastes by ion exchange only.

This plant is expected to be in operation in the summer of 1959.



PLAN
WASTE TREATMENT PLANT
TA-35

Figure 5

Laundry wastes.—All of the protective clothing used at Los Alamos is laundered at a central contaminated clothing laundry. The waste waters resulting from this operation contain some or all of the radionuclides used at Los Alamos as well as soaps, detergents, etc. Fortunately, the wastes are only slightly above drinking water tolerance and they are easily disposed of in a seepage and evaporation pit. Available water dilutes the wastes to well below acceptable levels.

Cost of operation of the three treatment plants are shown in table I. It should be pointed out that the major capital investment cost at the Ten site plant ($160,000) is for the waste storage tank.

TABLE I.—*Annual cost of liquid waste treatment operation, 1957*

	TA-45	DPW	Ten site
Labor, supervision, laboratory	$9,750	$11,280	$6,800
Miscellaneous equipment, supplies and utilities	3,340	2,750	1,750
Treatment chemicals	1,570	3,240	1,400
Maintenance	4,800	3,000	4,000
Depreciation* (see below)	20,750	11,500	17,200
Total annual cost	40,210	31,730	31,150
Total flow to plants, 1957 (gallons)	11,538,800	4,712,900	465,850
Unit cost (per gallon)	$0.0035	$0.0067	$0.067
Capital investment (chargeable to waste treatment), total	250,000	125,000	215,000
*Structures (depreciation at 5 percent a year)	85,000	20,000	87,000
*Equipment (depreciation at 10 percent a year)	165,000	105,000	128,000

A rather extensive environmental study program has been conducted at Los Alamos during the past 10 years to determine the effect of discharge of liquid radioactive waste on the environment. This program is jointly sponsored with the U.S. Geological Survey and includes sampling of soils, streams and well water. Six test wells were drilled in 1950 to intercept any radioactive waste which might reach the water table below the townsite. The water in these wells has been sampled and tested on a regular basis since that time and no detectable contamination has been found. Other environmental samples have, likewise, shown no appreciable movement of any radionuclides and indicate that the small amount of radioactivity discharged at Los Alamos has remained in place and well within the confines of the project.

As pointed out above plutonium and other radioactive isotopes are quite firmly held to the soils in the project. This fact has also been demonstrated at other A.E.C. sites. The decision to build the treatment plants was made despite these facts because it was felt that when the wastes are discharged to the ground all control is lost. It was deemed wise to get the material into an insoluble form before ground burial so as to maintain somewhat more control and to minimize translocation of the radioactivity.

AIR-BORNE WASTES

Because of the extreme hazard that plutonium represents in the air-borne state, it is necessary to perform most of the work with this element in hoods or dryboxes which are maintained under a negative pressure by exhausting the air. The resultant air carries plutonium dust which must be removed before the air is discharged through stacks.

The cleaning must be extremely efficient and particles of 0.3 to 1 micron size (.00007 to .0003 in.) must be removed. (By comparison, the smoke particles in cigarette smoke are in the order of 0.3 microns, and although some moisture and tars are held by the filters in cigarettes very little of the smoke itself is removed.) The toxicity of plutonium can best be shown by the fact that poisonous limits of concentration of this material in air are about 10 million times less than are the ordinary nonradioactive materials such as carbon monoxide. In absolute values the concentration must be reduced to less than one ounce distributed uniformly throughout 500 cubic miles of air. This amount of air would cover the area of Washington, D.C. (69 square miles) to a height of 7¼ miles.

DP west area.—This facility is engaged in the formation of plutonium metal from its salts, fabrication of the metal and recovery of the metal from various materials. The air-borne material in this area consists of plutonium oxides and salts which result from processing plutonium metal. The air requirement is 184,000 cubic feet per minute.

For some years an electrostatic precipitator was used in the air cleaning but due to high maintenance costs it was abandoned and the present system of dry

filter pads was installed. In this system the air is conducted through ducts and passed through filter pads which consists of 10-ply paper filters and are 24 inches by 24 inches by 8 inches in size. These filters will remove over 60 percent of the particles in the range of 0.3 to 1 micron and practically all of the material of larger size. Because all of the air to the plant is prefiltered (for product quality control purposes), the filters do not clog easily and are replaced only about three times per year. The used filters are taken to the contaminated dump for burial. The air, after filtration, is discharged through 50-foot-high stacks to the atmosphere.

CMR Building.—In the Chemical and Metallurgical Research Laboratory building the air is cleaned by a so-called wet type of air cleaner. Here the contaminated air results from exhausting the air from hoods and dryboxes in which physical, chemical and metallurgical tests are made on plutonium, uranium and other radioactive materials. The air requirement at this building is 350,000 cubic feet per minute.

The air is conducted through ducts and first passes through two wet pads which consist of glass fibers packed in a 20 inch by 20 inch by 8 inch cell. A photograph of the filter pad and sprays is shown in figure 6. These cells are sprayed with water and the air then passes through zigzag water eliminator plates before passing through dry filter pads similar to the wet pads but of finer material. The air then passes through another series of identical filters before being discharged to the atmosphere through 60 foot stacks. The wet and the dry filter pads are changed at intervals of from 1 to 2 years and the used pads are taken to the contaminated dump for burial. The water used in the wet pads is sent to the liquid waste disposal facility for treatment. An outside view of the complete installation is shown in figure 7.

Because of the high cost of operation and maintenance of this type of filter, the system hase been gradually converted to the dry-pad type similar to that used at DP West.

Ten Site.—The air-borne wastes here contain radioactive barium and lanthanum which are gamma emitters and the original installation was very similar to the system in the CMR building. The wet-type filtration method was used here so that the activity could be transported in water from the filter building to the liquid waste storage tanks so as to minify radiation exposure to plant operating personnel. However, the system did not operate exactly as expected and there was still a considerable amount of build-up of activity in the filters which presented a slight hazard to operating personnel. For this reason a dry-pad filter, buried underground and shielded with lead and concrete, was installed ahead of the wet-type filter. This has operated very satisfactorily over the past several months.

TA–48.—In the radiochemistry building which deals with a wide variety of radioactive isotopes including radioactive iodine and acid mist a venturi scrubber is used. In this installation a solution of 1 normal caustic is recirculated through the scrubber where it is intimately contacted with the air-borne wastes and absorbs the nuclides. This system has worked extremely well since its installation last year.

In addition to the three main installations described above there are a number of the small air cleaning facilities throughout the project. These generally utilize dry filter pads which are changed as the need arises.

Because the conversion from wet-type to dry-type filtration at the CMR building and at Ten Site has been done only recently, sufficient operating cost data are not available at this time. Cost data for air cleaning at DP West are summarized below:

	DP West (dry)
Volume of air in cubic feet per minute	184,000
Capital investment	$15,800
Depreciation[1]	790
Operation (labor and materials)	14,700
Maintenance	1,500
Total annual cost	16,990

[1] Based on life expectancy of 10 years.

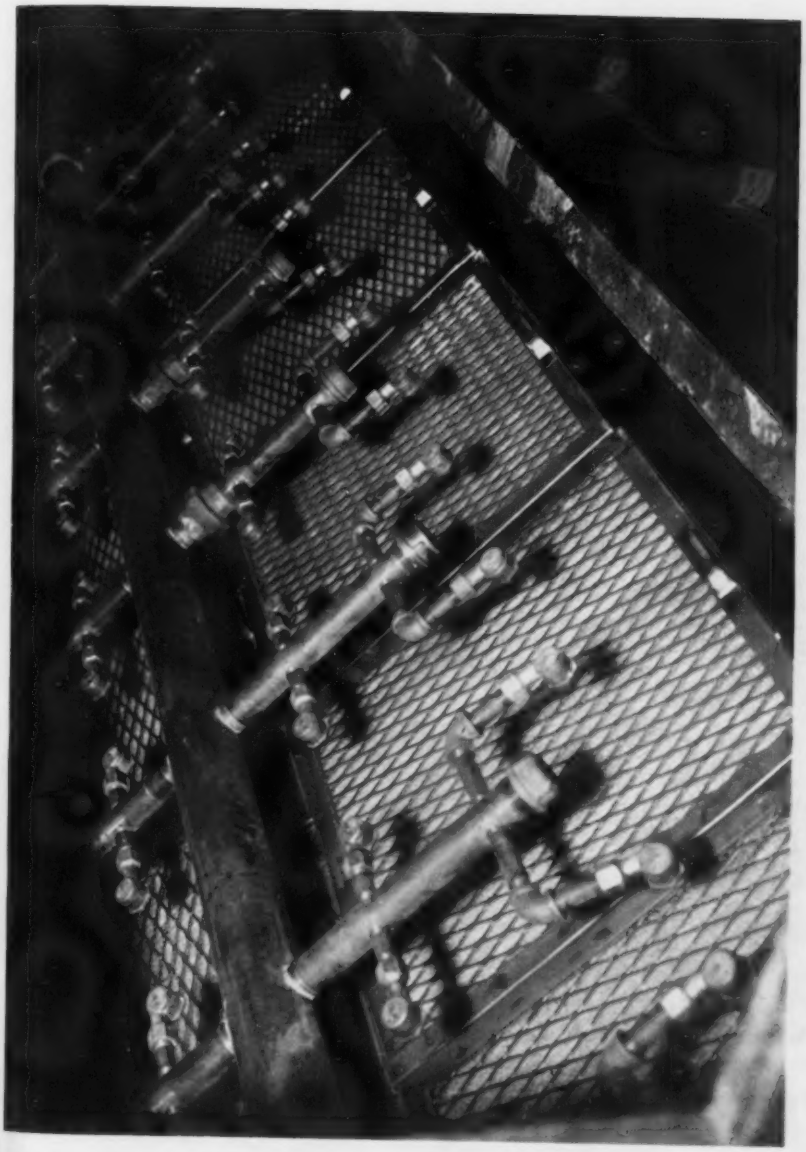


Figure 6

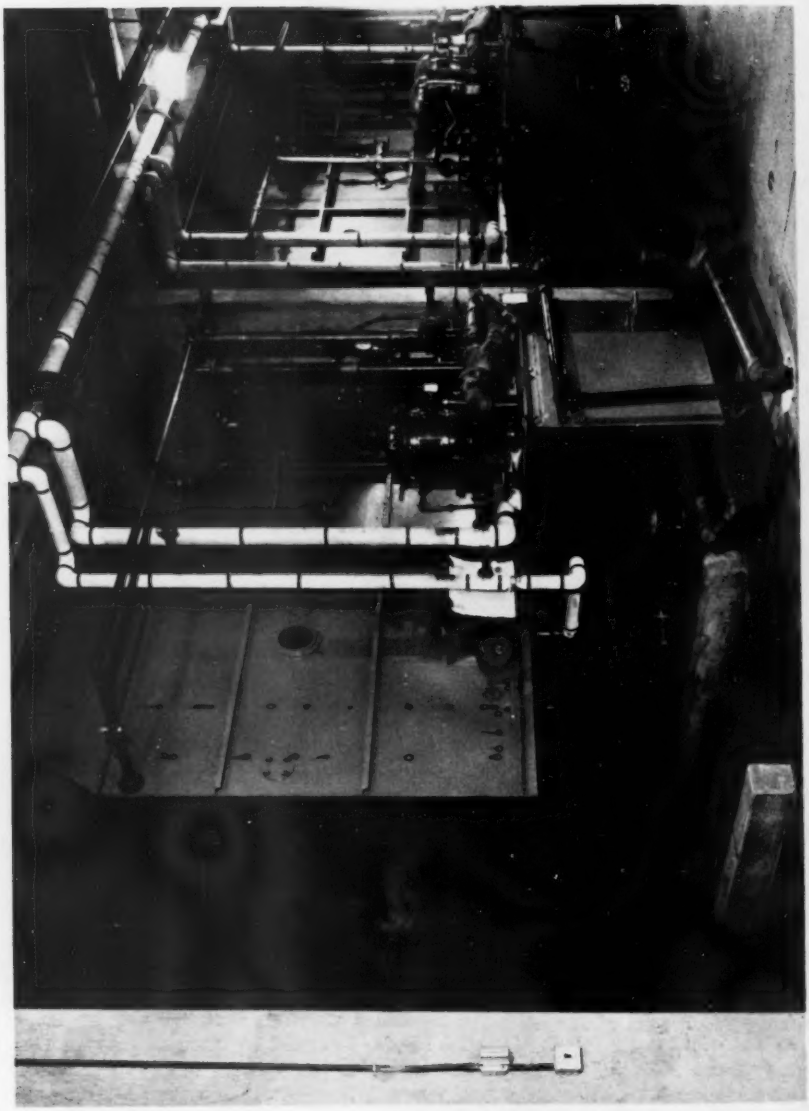


Figure 7

The cost of the air ducts for conducting air to the cleaning facilities has not been included in the above figures.

The operation of the various air cleaning facilities at Los Alamos has proved to be very satisfactory over the years. The air in areas surrounding the project has been monitored continuously and any occasional high value of radioactive materials in the air has been shown to be due to the result of nuclear bomb tests and has not been contributed by Los Alamos. Traps for catching fallout which might arise from stack discharges have repeatedly shown negative results. Efforts to improve efficiency of air cleaning and economy of operation are on a continuing basis.

SUMMARY

The ideal method of disposal of radioactive wastes would obviously include concentration to the smallest possible package and fixation of the small package in a perfectly insoluble and unleachable form. This philosophy has been practiced at Los Alamos as closely as possible. The radioactivity in the solid wastes is, naturally, quite firmly fixed before burial and the material is compacted quite thoroughly during burial. The sludge resulting from the treatment of liquid wastes are extremely insoluble and resist leaching. It is felt that this method is better than disposal into the ground in the liquid form where control of the wastes is lost. The air is cleaned to a point well below acceptable levels and the resultant filters, containing the radioactivity, are buried in well-controlled dumps.

The half-life of plutonium is in the order of 24,000 years so that the areas receiving these wastes are essentially condemned for all time. Admittedly, Los Alamos does not follow the ideal practices as pointed out above. However, it is felt that, because of the remoteness of the area as well as the limited value of the real estate, it is possible to utilize somewhat more land for burial than would be the case in some other areas in the United States. It should be emphasized, however, that a continuing research and development program is being carried out to improve the present methods of disposal.

Figure 7

II.

WASTE MANAGEMENT SAVANNAH RIVER PLANT ATOMIC ENERGY COMMISSION

PREPARED BY ROBERT J. CHRISTL E. I. DU PONT DE NEMOURS & COMPANY

I. INTRODUCTION

In the preparation of this paper it has been assumed that the relative abundance of radioactive materials and the physical effects of these both separately and in toto have been adequately covered in other hearings. The information presented, therefore, will be limited to the physical handling and disposition of radioactive wastes at the Savannah River Plant of the Atomic Energy Commission for which E. I. du Pont de Nemours and Company is the prime contractor.

Before presenting details concerning waste handling, it should be emphasized that the basic principle guiding operations at Savannah River is that of total containment. This principle is compromised only when the economics are substantially in favor of some other approach and when it can be assured that radioactivity at the site boundaries will be substantially below the maximum permissible concentration allowed by the National Bureau of Standards. In fact, every effort is made to maintain discharges at such levels that the normal background is not raised appreciably.

The Savannah River Plant occupies an area of about 320 square miles along the Savannah River downstream from Augusta, Georgia. The area is humid with about 50 inches annual rainfall and is located on the southern Atlantic Coastal Plain. The water table is relatively close to the surface (30-60 feet) and discharges into surface streams within the plant boundaries. The soil has relatively poor characteristics for the retention of wastes by ion exchange. The areas immediately adjacent to the site are not densely populated but there are several towns and cities within five to twenty miles of the boundaries. These factors combine to limit the possibilities of radioactive waste disposal to the environment and, when considered in the light of the inexact knowledge of the biological effects of the various isotopes, led to the adoption of the stated philosophy of substantially complete containment.

At the Savannah River Plant the primary potential sources of radioactivity release to the environment are the reactor and the chemical separations areas. Containment of activity in the reactor areas is favored by the use of indirect cooling of the reactors since river water is not exposed to intense neutron fluxes. In the cooling basins the irradiated metal is separated from the water by its relatively inert aluminum jacket. The separations areas, however, are faced with containment of the fission products after the irradiated metal is dissolved. Treatment of off-gas streams and liquid wastes has effectively contained these fission products so that most are stored in underground waste tanks with very minor release to the environment.

All of these operations are controlled by suitable analyses and by on-site monitoring. The results of these programs indicate that all off-site environments are maintained well below recognized limits for exposures. As an indication of the effort involved in containing wastes, the direct cost of environmental monitoring and operation of storage and disposal facilities is about $750,000 per year and the capital investment in waste storage is about $21,000,000.

The following paragraphs describe in more detail some of the factors outlined above.

II. GEOLOGY AND METEOROLOGY

During the site selection study and during design and construction, the available information relating to weather, geography, population and major installations was reviewed and additional studies undertaken where required, particularly to cover the more limited area involved as compared with the generally available regional information. A very detailed study of local climatology was undertaken to facilitate forecasting the dispersion of any activity released from the operations.

In general the Savannah River area is characterized by mild winters and warm, humid summers with a gradual transition between. On the average, rainfall is greatest in July and least in October; this condition results in the main from the greater thunderstorm activity during the summer period. The winds are variable in direction and exceed 3.5 mph about 85% of the time, a factor which is favorable to dispersing any materials vented to the air. The frequency of showers, however, is a disadvantage, since rain can be expected to remove a relatively large proportion of any contamination from the air. Normal releases, therefore, are limited to those which will cause no difficulty under the most adverse conditions.

The Savannah River Plant site is effectively drained by the Savannah River and tributary streams which empty into the river within the boundaries of the site. Because of the sandy surface, the runoff is practically negligible except after heavy rainfall, thus the streams are fed from the ground water with quite uniform flow.

The ground water occurs as perched water tables, normal water tables and artesian aquifers. The perched water tables result from clay lenses of varying and indefinite extent in the subsoil. The normal water table is generally encountered at a depth of 30 to 60 feet, lying deeper for the upland areas. There is insufficient data to predict with any assurance the direction and the flow of the ground water at a particular location on the site. The problem is complicated by the variable and undefined layers of fat clay in the clay sands and sandy clays. It is further complicated by the cavernous zone, 50 to 70 feet thick, which lies at a depth of 50 to 100 feet and slopes to the Savannah River. This zone has been shown to result from solution of calcareous materials.

Artesian water is obtained from the Tuscaloosa formation which lies about 240 feet below the upland surfaces and about 80 feet below the bed of the Savannah River. The static head reaches about 30 feet above ground at the Savannah River bottom, but is about 110 feet below ground in the upland areas.

The clays encountered are generally kaolinitic and have a low exchange capacity compared with the montmorillonites found at other sites. A limited amount of work has shown that the characteristics of the soil are such that it could be expected to hold some radioactivity for extended periods of time. Overall, the conditions appear to indicate that accidental spills will be retained to the extent that no danger to the public area will be encountered but that the soil should not be relied upon for the routine disposal of large quantities of activity.

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III. SOURCES OF WASTES AND THEIR DISPOSAL

The primary sources of activity are encountered in the reactor and separations areas with the latter presenting the major problems with respect to waste disposal. The control and research laboratories send liquid wastes to the separations areas for processing and solid wastes to on-site burial. The fuel fabrication areas generate small quantities of wastes with natural activity which is disposed of in seepage basins.

A. Reactors

The Savannah River reactors are heavy water moderated and cooled. The economics of this system dictate indirect heat exchange and thus alleviate problems related to the disposal of radioactivity since the cooling water from the river is not exposed to the neutron flux in the reactors and, therefore, does not build up induced radioactivity. The generalized scheme of a reactor area is shown in Figure 1. The fuel is jacketed with aluminum or other corrosion resistant metal to minimize the possibility of spreading fission products not only in the reactor but also during handling after discharge. As indicated previously, the moderator-coolant is cooled by river water in a shell and tube type of heat exchanger. The cost of the heavy water - dollars per pound - makes it imperative that losses be held to a minimum. For this reason the river cooling water is monitored to detect moderator leakage and, also, to ensure that radioactive materials accompanying a leak are not discharged directly to the river.

The moderator does pick up activity as the result of the formation of radioactive tritium by neutron bombardment and as a consequence of corrosion of irradiated metallic components including the fuel cladding, Suspended solids and dissolved metals are removed by a filter ion exchange system. Spent filters and resin beds are removed and transported to the burial ground. Since tritium is an isotopic containment of the moderator, its separation would be quite difficult. This implies that any leakage of moderator, as such, or moderator wetting the fuel during discharge tends to vaporize and thus contaminate the atmosphere. The reactor buildings were designed with adequate ventilation to dilute such activity and discharge it through a 200 foot stack. To date, there has been no occasion where the tritium concentration either on or off site has approached the permissible limit for continued exposure.

The irradiated fuel discharged from the reactor is removed to a storage basin where the short-lived isotopes are permitted to decay to minimize the problems encountered in further processing. Following decay cooling, extraneous housing tubes, etc., are removed, baled and sent to the burial ground. The clad fuel is shipped in shielded containers to the separations areas. Rupture of the cladding on a fuel element offers an additional potential source for the spread of contamination. A section of the storage basin is isolated to permit handling such elements. In actual practice they are inserted in a special sealable container as soon as possible. Whereas the normal flow of basin cooling water is directly to the river, water from the emergency basin can be diverted to a seepage basin if it becomes contaminated.

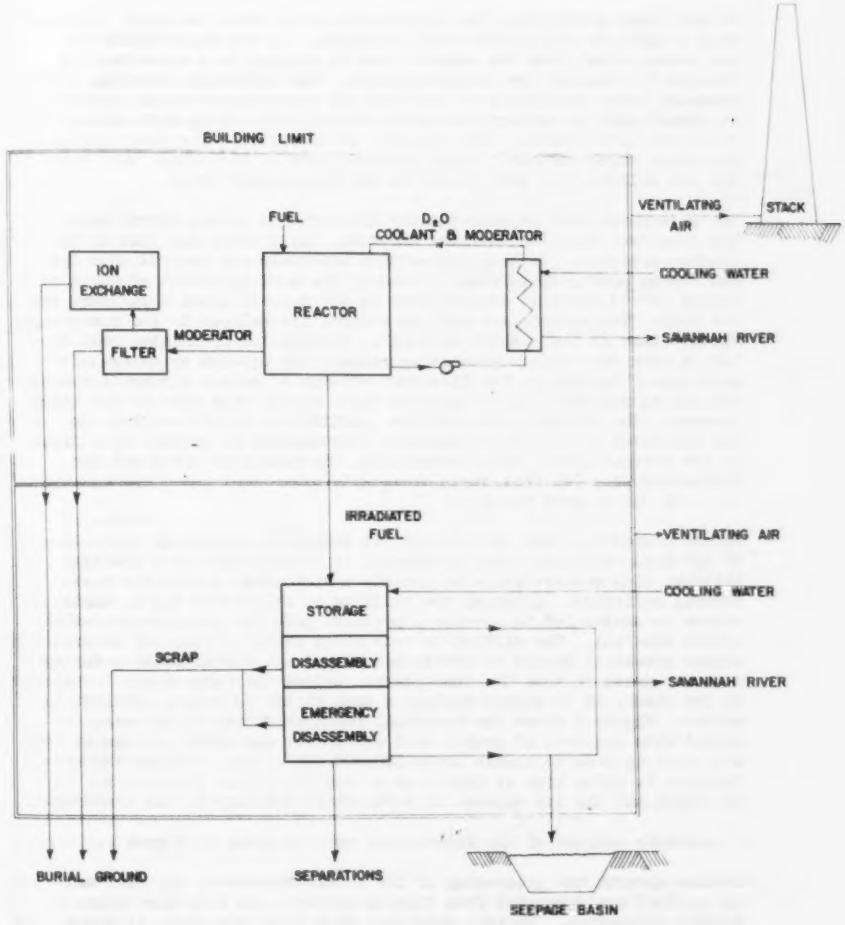


FIGURE 1

REACTOR AREA

B. Separations

As mentioned previously, the separations areas offer the major problems with respect to radioactive waste disposal. In the first operation, the cooled metal from the reactor area is charged to a dissolver and treated to dissolve the aluminum sheath. The resultant solution, which contains minor quantities of the fuel and associated fission products, is transferred to underground waste storage tanks along with certain other low level wastes. For purposes of definition, low level wastes are those which contain fission products only to the extent that they are not self-heating when stored in the underground tanks.

The dejacketed fuel is subsequently dissolved in strong nitric acid. The resultant solution contains the fuel, the product and most of the fission products. Certain radioactive materials are removed with the gas stream during dissolving. Probably the most important of these is iodine (I^{131}) since it concentrates in the thyroid gland when taken into the body. Two methods are used to control its release to the atmosphere. First, since it has a short half-life, cooling the irradiated fuel for 120 or more days before processing reduces the problem by orders-of-magnitude. Currently, the dissolver off-gas is passed through a reactor containing supported silver nitrate which reacts with most of the iodine present. The off-gas also contains particulate matter carried out of the dissolver or the silver reactor. Particulate is removed by a filter in the off-gas line. After processing, the dissolver off-gases are discharged to a 200 foot stack along with other vent gases and exhaust air from the process building.

Before proceeding with the process, it should be emphasized that all of the high radiation level processing is accomplished in a concrete building with massive walls to provide a biological shield for penetrating radiation. Although the building is relatively tight, ventilation is controlled to provide a positive path for any airborne radioactive material. The airflow is from clean areas to those of successively higher potential levels of contamination and all contaminated areas are at lower pressure than the atmosphere. Before the exhaust air is released to the stack, it is passed through a sand filter to remove particulate matter. Figure 3 shows the essential features of the filter which is packed with six feet of graded sand and gravel and which is roughly one-half acre in area to handle about 100,000 CFM of air. Filtration efficiency is quite high as indicated by samples of the filtered air to the stack and the low release of particulate activity to the atmosphere.

A schematic outline of the separations area is shown in Figure 2.

Continuing with the processing of the irradiated metal, the fuel and the product are separated from fission products and from each other by solvent extraction. Various chemicals such as nitric acid, aluminum nitrate, ferrous sulfamate, sodium nitrite and sodium hydroxide are employed as salting agents, for valence adjustment and for solvent treatment to reduce activity and remove degradation products caused by chemical and radiation damage. All of these tend to accumulate in the waste streams containing the bulk of the radioactive fission products.

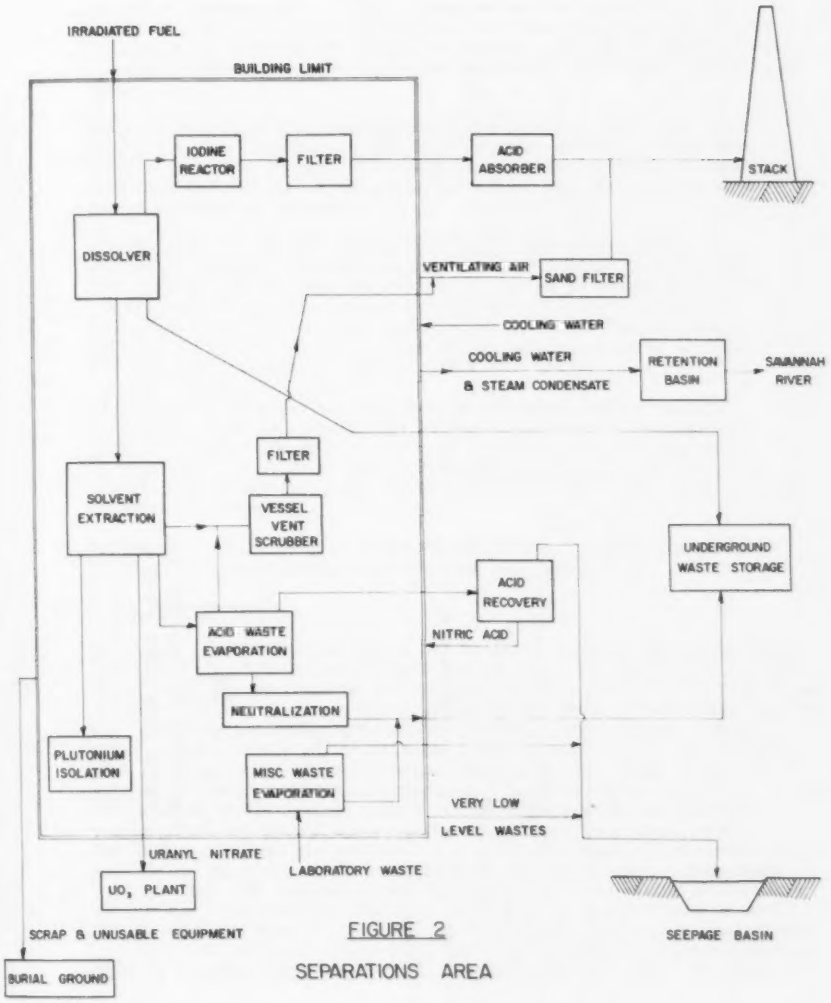


FIGURE 2

SEPARATIONS AREA

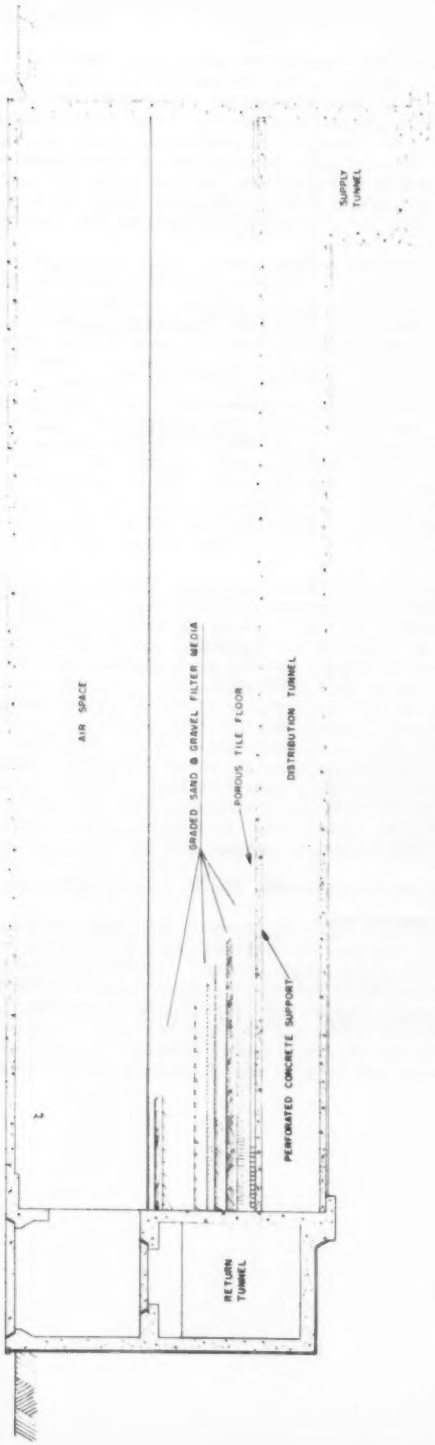


FIGURE 3

SAND FILTER FOR EXHAUST AIR
FILTER AREA 100FT. X 240FT.

To minimize the volume of material to be stored, all waste streams are evaporated to the maximum solids concentration possible without the formation of precipitates which would tend to plug the lines to the waste storage area. Waste streams containing appreciable quantities of nitric acid are evaporated before neutralization to recover acid and avoid bulking the streams to storage with inert sodium nitrate. All liquid wastes are neutralized and stored in underground carbon steel tanks.

The acid recovered from waste evaporation is concentrated for reuse in the process. Condensate from acid recovery as well as that from other evaporation is sent to seepage basins after analysis to determine that the radioactivity levels are within the limits prescribed for this method of disposal.

All equipment in the processing building is positively vented to a collection system to prevent the uncontrolled release of activity. The contaminated air from the collection system is passed through a scrubber to collect particulate matter, aerosals, iodine and any ruthenium which may have been volatilized as RuO_4. The scrubber is followed by a separate fiber-glass filter before the gases are mixed with the main exhaust air stream to the sand filter.

Cooling water from the various vessels and steam condensate from evaporator coils is normally not contaminated and is discharged to surface streams. To avoid release of activity, however, the water is monitored and flows through a delaying basin which has sufficient capacity to permit shutting down the operation before activity actually reaches the streams.

Radioactive solid scrap is packaged and buried. Process equipment which is not economically repairable is also buried.

C. Waste Storage Tanks

As stated previously, the policy for handling wastes at Savannah River Plant is that of total containment and of reserving the natural retention capacity of the environment for emergencies. In view of this, the design criteria for the waste storage area called for maximum tank security without, however, incurring unreasonable capital costs. Figure 4 shows a cross-section of one of the original waste storage tanks. The primary tank is constructed of carbon steel with all welded seams being radiographed to assure quality construction. The tank rests in a steel "saucer" designed to retain leakage from a faulty primary tank, at least for a period until the stored waste can be transferred to a sound vessel. The tank and saucer are enclosed in a reinforced concrete structure with an annular space to permit inspection of the tank proper. Tank vents are provided with condensers and filters to minimize spread of contamination due to breathing. Further, cooling coils are provided to avoid the possibility that the radioactive decay heat will lead to violent and uncontrolled boiling. One problem has been encountered in waste storage that is somewhat more severe than originally contemplated. The high

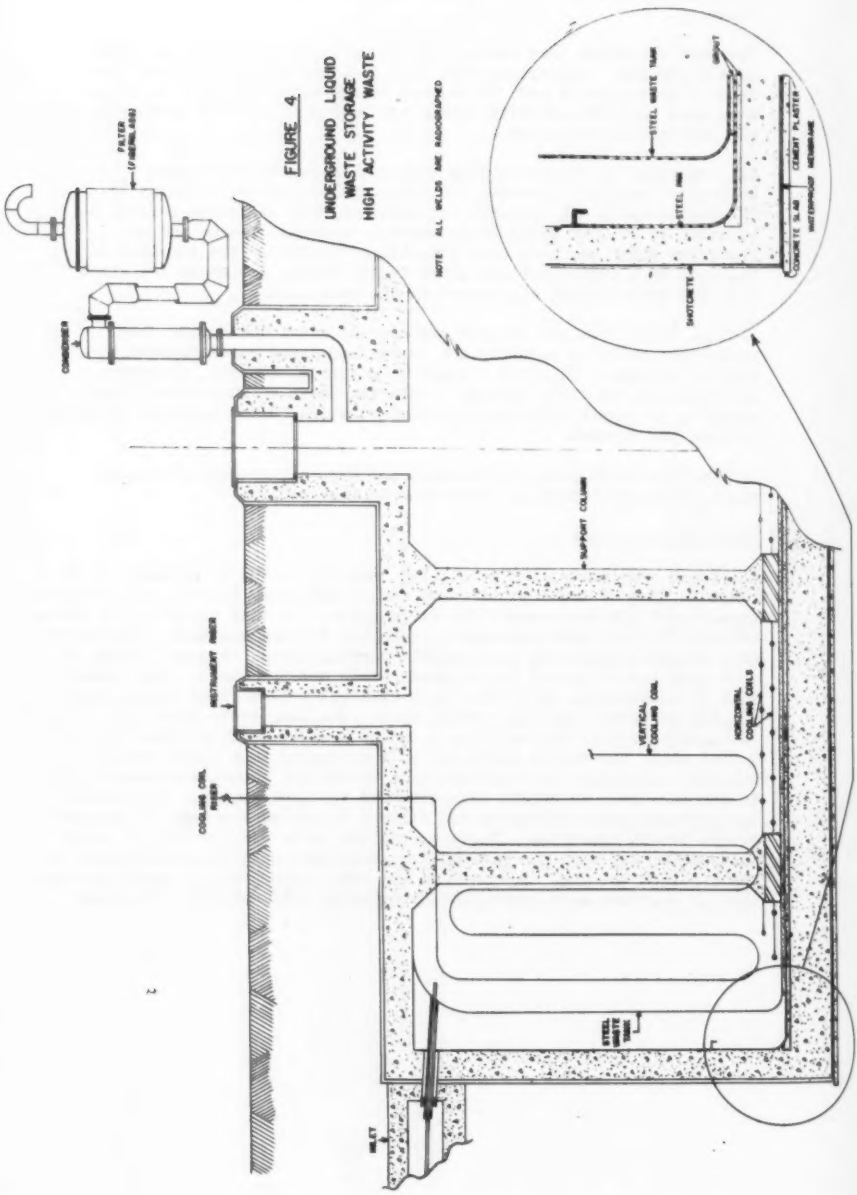


FIGURE 4
UNDERGROUND LIQUID WASTE STORAGE HIGH ACTIVITY WASTE
NOTE ALL WELDS ARE RADIOGRAPHED
FILTER
CONDENSER
INSTRUMENT RISER
COOLING COIL RISER
SUPPORT COLUMN
VERTICAL COOLING COIL
HORIZONTAL COOLING COILS
STEEL WASTE TANK
INLET
STEEL WASTE TANK
GROUT
STEEL PAN
CEMENT PLASTER
CONCRETE SLAB
WATERPROOF MEMBRANE
SHOTCRETE

level wastes generate a precipitate which carries an estimated 90% of the fission products with the result that the bottoms of the tanks have heated to about 300°F although the supernate was cool.

It was realized that the tank selected was relatively expensive on a unit cost basis. When it became necessary to build additional waste storages, a comprehensive design study was undertaken which resulted in about a 30% reduction in the unit cost of tankage. One significant change in philosophy was the use of a pump in the transfer of wastes rather than the previous reliance entirely on gravity. This significantly reduced the costs of excavation and waterproofing.

As operating experience was gained and the characteristics of the waste streams became better defined it became possible to reconsider the ground rules for waste tank construction. As a result, a further increment of waste storage was constructed having the essential features shown in Figure 5. The tank as shown is intended to be used on low level wastes or high level wastes which have cooled to the extent that they will generate no more heat than will be lost to the ground. These tanks are still high integrity storage with the steel liners being radiographed and the concrete shell being prestressed to minimize the possibility of cracking. On a unit cost basis, however, these tanks cost only about 25% of the original complement.

In addition to savings on unit costs of tankage it should be emphasized that appreciable savings in waste storage costs have resulted from process development. For example, acid recovery from high level wastes reduced high activity waste volume about 50%. Other reductions have been possible by rebalancing extraction conditions to reduce the quantity of non-volatile additives.

D. Burial Ground

All Savannah River Plant solid wastes are buried in a centrally located fenced area. Most burial is in slit trenches which are backfilled as waste accumulates. A shielded crane is available to handle heavier loads having high radiation levels. All activities are monitored by health physics personnel.

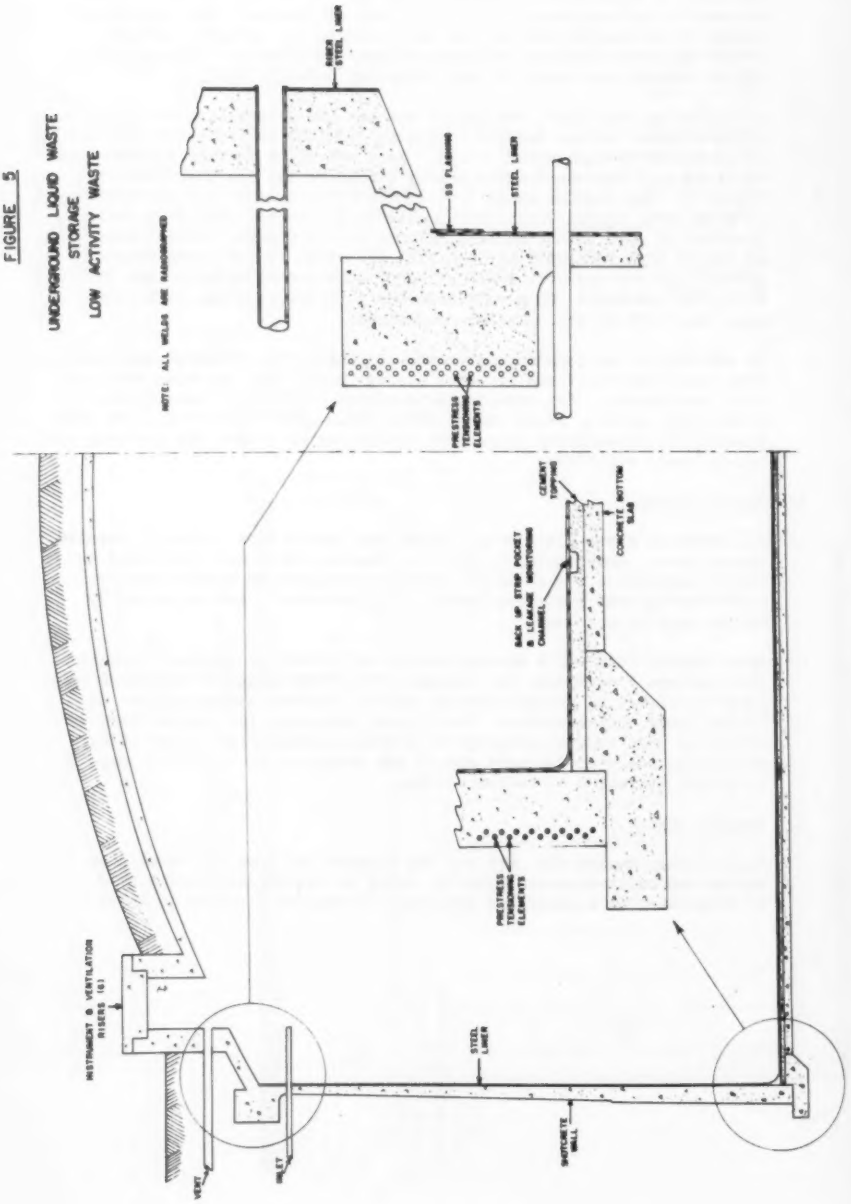
As a general rule solid materials such as laboratory wastes, cleaning pads and small equipment are wrapped to prevent spread of contamination. Large equipment is normally washed and/or decontaminated so there is little loose contamination. Where loose contamination exists every effort is made to seal openings to prevent contact with ground water. Monitoring wells are located around the perimeter of the burial ground to detect migration of radioactivity.

E. Seepage Basins

Open seepage basins are used for the disposal of very low level high volume radioactive wastes since it would be economically unfeasible to evaporate the hundreds of thousands of gallons involved to store

FIGURE 5

UNDERGROUND LIQUID WASTE
STORAGE
LOW ACTIVITY WASTE



the very small amount of radioactive materials in the waste tanks. Although all areas have seepage basins, only the separations areas utilize them for the routine daily disposal of contaminated waste.

Each separations area has a series of three basins with a total area of about five acres and a depth ranging from six to ten feet. In each series the first two basins are small in area and were intended primarily as settling basins to leave the third relatively free of solids and maintain good seepage rates for a long period. Flow to the basin averages about 80,000 gallons per day and the total activity to the basins to date has been 2.5 curies of alpha emitters, 240 curies of non-volatile beta emitters and 2300 curies of I^{131}.

Ground water around the basins has been monitored by use of a total of 25 steel-cased permanent wells and 75 temporary uncased wells. As might be expected from the variable geology of the site, the results obtained on samples from the various wells have been variable. Samples from all wells near the basin show radioactivity which can be traced back to the basins. In all cases, however, the concentration of activity in the normal water table is lower than the maximum permissible concentration in less than 100 feet from the basins and before there is any flow to surface streams.

IV. ENVIRONMENTAL MONITORING

An extensive program is in force to assure that Savannah River Plant practices do not contribute significantly to radioactivity in the public zone. This effectively cross-checks operating limits imposed on the various operations. For example, the total iodine release from the separations areas stacks is limited to 124 curies per month and the total β-γ to each area seepage basin is limited to 90 curies per month. Organizationally, the monitor and control of working conditions with respect to personnel exposure as well as area monitoring is independent of the technical and production organization. This arrangement is a deliberate effort to maintain complete objectivity in the interpretation of radioactive risks both on and off plant. The Savannah River is sampled above the plant for reference and then at several points as it passes the plant to determine the contribution of the various streams and finally downstream to obtain the total Savannah River Plant contribution. Air monitoring stations are located off-site to obtain representative information around the entire periphery of the Savannah River Plant. In addition, samples of vegetation and aquatic life are evaluated. A total of 25,000 samples were evaluated in this program during 1957. Typical information collected is as follows:

Activity Released from Separations Area Stacks, Curies

	Alpha	Nonvolatile Beta	Radioiodine
1955	2.90	33.9	69
1956	0.15	8.4	1,576
1957	0.05	1.7	292
	3.10	44.0	1,937

It can be seen that the alpha and beta emissions were reduced. This is attributable to changes in equipment or process after the source of loss was identified. The high I^{131} release for 1956 was the result of a plant test to determine minimum cooling times for irradiated fuel that could be tolerated. The effect of I^{131} releases on the atmosphere is shown in the following table; units are 1.0×10^{-14} μc/cc.

	Separations Areas	Other SRP Monitoring Stations	Off-Plant Monitoring
7/55-12/55	18	4	3
1956	580	50	15
1957	186	43	36

The effect of Savannah River Plant release on the Savannah River is shown by the following average stream analyses for nonvolatile beta expressed as 10^{-15} μc/ml.

	Up River Control Highway U.S. 1	Upper Three Runs	Four Mile Creek	Steel Creek	Lower Three Runs	Downstream from Plant Highway U.S. 301
1953	-	7	7	7	7	-
1954	16	8	9	9	9	10
1955	10	16	10	12	12	13
1956	12	13	16	22	14	14
1957	17	17	20	56	35	34

The increase in background from 7 to 17 x 10^{-15} c/ml owing to bomb tests is apparent as is the minor contribution from Savannah River Plant except for 1957. The increase for 1957 is entirely attributable to a single incident during the year associated with experimental work.

Since the selection of the Savannah River site there has been continuing review of our plant design philosophy and plant operation as related to environmental effects with the U. S. Department of Public Health and the Health Departments of both Georgia and South Carolina. This liaison has been made easy by formation in 1951 of the Savannah River Advisory Board composed of representatives from the several government agencies and the contractor. In addition, the Academy of Natural Sciences was employed for a survey of aquatic life prior to the start of plant operations and now performs routine checks to determine the effect of plant operations on the Savannah River.

V. ECONOMIC ASPECTS

The large scale handling of radioactive wastes with proper regard to the safety of employees as well as the general public involves appreciable costs. Every effort is made to effect economies as experience and development permit. Direct annual cost related to the Savannah River Plant activity is about $750,000 per year.

In addition there are capital charges for the various facilities. The waste storage areas, however, represent the major expenditures. The following are volumes and costs of the waste storage tanks at the Savannah River Plant.

	No. Tanks	Total Gallons	Cost
Original Plant	12	9,000,000	$14,200,000
Added 1956	4	4,100,000	4,700,000
Under Construction	4	5,200,000	2,300,000
	20	18,300,000	21,200,000

As an illustration of Savannah River Plant progress in reducing the costs related to waste storage, the following volumes and costs apply to processing a quantity of uranium which yielded 1000 gallons of total waste for storage soon after plant start-up.

Date	Waste Vol. Gals.	Tank Cost $/Gal.	Storage Cost $/Unit of U
1/55	1000	$1.60	$1,600
1/56	670	1.60	1,035
3/56	375(1)	1.60	600
1/58	220	1.15	255
10/58 (Est.)	*115	1.15	180
	**105(2)	0.44	
7/59 (Est.)	*115	1.15	145
	** 35(3)	0.44	
Final	75	0.44	35

*Volume of high activity waste
**Volume of low activity waste

Notes (1) Started acid recovery from high activity waste.
(2) Tanks now under construction available for storage of low level waste.
(3) Start evaporation of low level wastes.

Note (3) refers to the future evaporation of low level waste. Test work has been completed on a prototype evaporator and construction has been started on a plant installation. It is anticipated that wastes now stored at 30-35% solids concentration can be evaporated to about 70% solids. The resultant concentrate will solidify as it cools to ambient temperatures. The "Final" condition listed in the table reflects the evaporation of high level wastes after they have decayed sufficiently that, following evaporation, heat release will not exceed that lost to the ground so there is no possibility of self-heating to temperatures which could jeopardize the tank structure.

VI. FUTURE DEVELOPMENTS

It is felt that the last described situation, with wastes solidified in a tank of high integrity but relatively low cost, is one which offers little or no concern for release of activity to the environment for the indefinite future. Even with failure of the steel tank liner and minor cracking of the concrete shell, it is postulated that water movement through the structure would carry only minor amounts of activity, (1) because the tanks are above the main aquifer and (2) the seepage would be equivalent to the annual rainfall, 45-50 inches, over the area of the fault.

Efforts will continue to reduce waste volumes by process improvement and progress in various approaches toward "ultimate" disposal will be followed.

A limited amount of work is being done on the relatively "cold" coating wastes, for example, since only a relatively low value of decontamination is required before this stream could be disposed of by seepage. Investigation of the Savannah River Plant site and properties of its soils will be continued to ensure that any deliberate disposal to the ground will be without risk to the uncontrolled area.

RADIOACTIVE WASTE DISPOSAL AT LAWRENCE RADIATION LABORATORY

(By Nelson B. Garden [1])

The problem of radioactive waste disposal at Lawrence Radiation Laboratory has been handled during the last 13 years through analysis and evaluation of the five basic factors into which this complex question has been divided:

1. Foremost is the introduction of the consideration of control and planning in regard to the handling of radioactive waste in the original concept of design construction and in specifications for operating procedures.
2. An acceptance of that quantity of waste which is deemed absolutely inescapable under No. 1 and an evaluation of it into either—
   (*a*) High level;
   (*b*) Intermediate or industrial level;
   (*c*) Low level.
3. Collection and processing or packaging for disposal.
4. Actual disposal—sea or land.
5. Economic considerations of the above.

No. 1.—Because of psychology, No. 1 has been the most difficult to cope with but when successfully carried out it is by far the most productive of efficiency, safety, and economy in dealing with radioactive waste disposal problems. This will be discussed briefly after the simpler items 2, 3, and 4 are covered.

No. 2(a).—High level—where radioactive waste exists in multicuries per liter in quantities (100,000 gallons or more) which makes it impractical to transport, it is considered high level. This limits the high-level problems largely to sites processing fuel elements. Proposals for handling this are quite numerous and after vast investigation and a multitude of reports no generally accepted system for disposing of this high-level waste has been selected. Lawrence Radiation Laboratory has participated in meetings to consider this problem but it will not be analyzed in detail here since this does not involve us directly at this time.

No. 2(b).—Intermediate level or industrial radioactive waste is readily identifiable by the criteria that even though the waste may be in the multicurie level, if the original radioactive material has been shipped to an industrial or other user the waste therefrom constitutes intermediate level waste. The fact that it has been shipped proves that the material can be put in a practical and adequately safe container for transportation and hence can likewise be put in a practical disposal container.

No. 2(c).—Low level radioactive waste is made up of millicurie quantities as contaminants or in solution and there is general agreement and acceptance of this.

No. 3.—As mentioned above processing or packaging the high level waste of the several production plants is considered a larger and special problem not to be dealt with here. An extension of the methods used for intermediate levels might lead to surprising results if applied to high level.

The basic philosophy at Lawrence Radiation Laboratory is to keep radioactivity under control at all times. To accomplish this it has been necessary to develop techniques and designs to maintain all operations in enclosures. The air contaminants are collected by filtration at the enclosure and the dry and liquid wastes are collected in batches at the enclosure. There are no hot drains and no hot ducts for air discharge to the atmosphere. A minimum of hold-up tank capacity (10,000 gallons) is provided for emergency only. This would be used to insure that an acid sewer contaminating incident did not

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[1] Born Apr. 29, 1900, in Pennsylvania.
Graduated from Cornell University with degree of electrical engineer, 1922.
Seven years in industry.
Graduate work at University of California with M.S. in chemistry, 1932.
Past 16 years with Radiation Laboratory, of which the last 13 years have been devoted to all aspects of processing and handling radioactivity, including the disposal of radioactive waste.

empty into the public sewer which adjoins the laboratory at a short distance. Should this occur (and it has not been necessary to use the tanks since their installation several years ago) the contaminated volume would be pumped out or gravity fed into concrete mixers to form blocks.

The collections from the enclosures thus are intermediate or low-level wastes and evaporation and incineration have both been considered in hopes they would achieve an advantage through concentration. Analysis at Lawrence Radiation Laboratory as at many other locations showed these attempts at concentration to be of questionable advantage especially if specifications under No. 1 are carried out efficiently and dilution held to a minimum. A gain might be indicated in a small percentage of the waste to be disposed of and these techniques might appear to offer advantages but ultimately the radioactive waste residue must constitute some form of waste disposal package and a careful and complete analysis is necessary before drawing economic conclusions.

As followed at Lawrence Radiation Laboratory the intermediate level waste can be incorporated in concrete either in small batches or in 10 or 20 ton concrete blocks as the circumstances dictate. This immediately puts an end to the packaging problem and the hazards of storage. Thus safe and convenient storage can be achieved while awaiting the choice of the most desirable and cheapest disposal location.

Where low-level radioactive waste is definitely known to be of short half life material, this is sometimes segregated and allowed to decay, but frequently the cost of the extra manpower and techniques will exceed the cost of processing the same material in the stream of waste being incorporated in concrete.

No. 4.—The disposal at sea or on land after the waste has been incorporated in concrete is an economic problem. Actual tests indicate that hazards are not likely to be created through the escape of radioactivity from a properly formed concrete block whether disposed of at sea or on land.

If either sea or land disposal of radioactive waste without packaging is considered, the problems of hazard, public opinion, and economics become very complex. If an empty salt mine or some other such void existed immediately under a processing plant it might be an advantage to use this for high-level waste disposal but if the solution must be transported to an underground dumping site the overall expenses chargeable to such transportation could conceivably result in a cost greater than putting the solution in concrete blocks and setting it aside in a storage area safely and for an indefinite time.

No. 5.—The economy considerations at Lawrence Radiation Laboratory begin with the original planning mentioned as No. 1. By control of dilution factors in air and process liquids the waste can sometimes be reduced by orders of magnitude. The known radioactive waste and also the suspected waste ore incorporated into concrete in as efficient a mixing arrangement as is justified. It is much cheaper to package the suspected waste than to employ people, equipment, and space to try and eliminate that portion which is not contaminated.

Located within several hundred yards of densely populated areas, the Lawrence Radiation Laboratory has successfully processed and disposed of kilocurie quantities of activity. Approximately 20,000 barrels of concreted waste have been disposed of at sea over the past 13 years and a recent survey [2] of the sea area (at 1,000 fathoms depth) failed to show any easily detectable contamination of the water, sea bottom or sea life and it may be that ultimate tests will show that nothing can be detected.

The technical details and the cost analysis for the waste disposal operations at Lawrence Radiation Laboratory are given in a recent report [3] attached hereto as an appendix. Also pertinent information has been compiled in a report [4] comparing the costs of land and sea disposal for Lawrence Radiation Laboratory.

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[2] Scripps Institution of Oceanography (April 1957).
[3] Lawrence Radiation Laboratory, UCRL–8380.
[4] Lawrence Radiation Laboratory, UCRL–8609.

THE DISPOSAL OF RADIOACTIVE WASTE MATERIALS AT THE UNIVERSITY OF CALIFORNIA RADIATION LABORATORY

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Abstract

The philosophy of "concentrate and confine" as applied to work with radioactive materials at UCRL has made possible special packaging and disposal procedures for radioactive wastes. These procedures and their advantages and disadvantages are discussed, and approximate costs are given for the various types of packages produced.

The disposal of radioactive waste materials is a major problem facing any installation processing or producing radioactive materials. The University of California Radiation Laboratory at Berkeley is engaged in nuclear research involving all types of radioisotopes, either produced by bombardment on its particle accelerators or obtained from the Atomic Energy Commission or commercial suppliers. There have been problems involving most of the isotopes from element No. 1 to element No. 102, in quantities from tracer levels to kilocuries, depending on the work and the availability of the isotope.

The Health Chemistry Department is charged with the safe handling of radioactive isotopes involved in laboratory research. The philosophy of the department is to keep radioactive materials as confined and concentrated as possible.

One result of the "concentrate and confine" method is that the volume of radioactive waste produced is kept small by avoidance of unnecessary dilution during the processing of radioactive material. Advantages resulting from keeping the volume of waste small include these:

- Less time and labor are needed to process the waste for disposal.
- The small volume of waste can be moved and processed not only more easily but also more safely, because safety features can be incorporated in the small containers that would make large ones very unwieldy.
- Less area and equipment are needed for processing and storage of packaged wastes prior to sea disposal.
- More packages can be shipped to the disposal barge at one time (so that fewer disposal shipments are necessary).

All these savings in time and effort amount to a gratifying economy: as is shown below, the disposal cost per "potential gallon" of liquid waste is less than 1½ cents.

About 200 researchers are continuously carrying on work at millicurie levels or higher in various departments at UCRL. These people depend directly on the waste-disposal services of the Health Chemistry Department. The average waste contribution is about 20 ft^3 of solid waste and 5 gallons of liquid waste per researcher per year. The wastes are packaged as described below for ultimate disposal at sea.

II. The Handling of Radioactive Waste Materials

The wastes produced throughout UCRL include the usual solid, liquid, and airborne types. All nongaseous waste is packaged in concrete. Liquid and airborne wastes (except for short-lived gases that may safely be released to the atmosphere) are solidified or trapped for incorporation in concrete packages. The concrete packages are made so that they can be stored safely for an indefinite period and eventually be safely transported by truck to a barge for disposal at sea at a depth of 1000 fathoms. (Section II D describes the types of waste packages used by UCRL.)

A. Solid Materials

These wastes fall into three categories, defined and handled as follows:

1. Contaminated materials from operations in special enclosures (gloved boxes, manipulator boxes), from animal injections, and from tracer-level work involving counting disks, etc.

The wastes in this category include contaminated glassware, rubber gloves, wipes, and other items removed from the enclosure during the operation or during the decontamination of the enclosure after the experiment. These wastes are collected in multilayer paper sacks and taken to the disposal area where they are incorporated in concrete in 55-gallon metal drums.

The solid wastes from high-level operations in special enclosures in the 6-inch lead cave are discharged into shielded waste containers located underneath the operating enclosure and connected to it by an airtight plastic tube 6 inches in diameter. After the operation the plastic tube is closed by a weld across the tube, the weld is cut, and the waste container with its own mobile lead shield is removed to the waste-processing area. There the waste

container is transferred from its lead shield into a specially designed concrete block for sea disposal. The lead shield is returned for further use.

2. Large contaminated equipment that would involve risk and (or) expense beyond justification during decontamination

Examples of items in this category are enclosures that have become so contaminated or corroded as to preclude recovery of the item by decontamination. Such large items are incorporated in concrete blocks measuring about 4 × 6 × 7 ft. As many as four standard enclosures can be disposed of in one of these blocks.

3. Equipment with internal contamination (i.e., the interiors of the metal parts have become radioactive owing to transmutation from accelerator bombardment)

This category includes parts of accelerators and other bulky equipment which cannot be reused or consigned to scrap owing to its internal activity. Such materials can serve as ballast in waste packages or sometimes can be dumped directly from the barge.

B. Liquid Materials

These wastes are classified according to activity level and chemical content.

1. High-level small-volume aqueous wastes (500 ml or less)

As these wastes develop, they are accumulated in polyethylene containers inside the work enclosure during the experiment. When work is completed these wastes are solidified by the addition of a mixture of Portland cement and vermiculite. Either the solidification is done in the enclosure or the polyethylene bottles are removed in sealed plastic containers to a special box used exclusively for the solidification of small volumes of liquid waste. The solidified wastes are then handled for disposal as small items of solid wastes.

2. High-level large-volume aqueous wastes (1 to 50 liters)

These wastes, produced during high-level chemistry work in the 6-inch lead cave, are discharged into specially designed and shielded waste containers. These containers (as for cave solid wastes) are stored behind the shielding and are connected to the cave box by means of plastic tubing.

After the experiment is completed the plastic connecting tubing is sealed off, then cut, and the shielded liquid-waste container is removed to the waste-disposal area. The wastes are then solidified either in the waste container or in a specially-constructed concrete block. (See Section II D 3 b.)

Other high-level large-volume wastes (such as scrubber liquors) are handled in essentially the same way.

3. Low-level large-volume aqueous wastes

These wastes are accumulated in the various laboratories in 5-gallon glass carboys. Whenever it is considered advisable, the carboys are taken to the waste disposal area where the liquids are pumped into a specially prepared 30-gallon steel drum. About 15 gallons of waste are solidified in one drum by the addition of cement and vermiculite.

4. Water-immiscible organic wastes

These wastes are handled in two different ways. Small quantities can be mixed with large quantities of the aqueous waste (up to 3% organics by volume). The mixture is then solidified by the cement-vermiculite process. The other procedure used is to solidify these organic wastes with "Napalm."

5. Nonradioactive high-volume laboratory sink wastes

The techniques and regulations for work with radioactive isotopes at UCRL under the "concentrate and confine" philosophy prevent gross contamination of the general laboratory sewer system. Nevertheless, good safety practices (and good public relations) dictate a check of these wastes for accidental contamination with radioactive materials. Toward this end, all laborabory sink wastes drain from the building through an acid waste system separate from the ordinary sanitary sewer. Though not necessarily noxious, wastes such as cooling water and aspirator discharges run through this system. Before these "acid wastes" are combined with the sanitary sewage they pass through a sampling system. This sampling system removes one-half gallon for every hundred gallons passing through the system. The half-gallon samples are discharged into a tank from which a daily sample is taken for analysis. In this way a historical record of any radioactive contaminant would be available should a contaminant be discharged accidentally into the sewer. Provision has also been made for an 8-hour diversion of building discharge to holdup tanks should an emergency arise.

C. Airborne Materials

These waste materials can be classified as solid, liquid, and gaseous products.

1. Solid aerosols (solid airborne waste materials)

Solid aerosols may be produced during mechanical operations (grinding of powders, machining, sparking, drying of solutions), or during chemical operations (such as smoke-producing reactions).

The disposal of these radioactive solid airborne wastes is dependent upon the type of air-cleaning mechanism used to remove these particulates from the airstream. Several types of air-cleaning devices are used at UCRL. For the standard type of enclosure (gloved box) a boxed filter is used. These are internally loaded filters of various sizes (usually $9 \times 9 \times 10\frac{1}{2}$ in.) and composition and are attached to the enclosure with 3 feet of 2-in. i.d. flexible duct-work. Exhaust air flow from the box through the filter is maintained by a blower or a manifold system attached to the outflow side of the filter. Planning for the particular service needs for an experiment permits scheduling of replacement of the filter as it becomes clogged or highly contaminated. The clogged or contaminated filter is incorporated as solid waste in concrete in a 55-gallon steel drum.

If a liquid-type air scrubber is used in the air-cleaning system of an enclosure, the resulting contaminated scrubber liquors are collected in polyethylene carboys. These liquors are then solidified as liquid wastes by the addition of the cement and vermiculite mixture.

2. Liquid aerosols (liquid airborne waste materials)

Air-suspended liquid waste materials resulting from chemical or physical operations (boiling, gas bubbling through solutions, etc.) are either trapped in de-entrainment devices and collected as liquid waste, or if small in quantity or intermittent in production are permitted to be trapped and dried on the normal air filter. The liquid collected is treated as liquid waste.

3. Gaseous radioactive wastes

Certain radioactive gases are allowed to escape to the atmosphere. Most of these are short-lived, and calculations combined with common sense determine the extent of precaution necessary.

An exception is the case of fission gases arising from the dissolution of irradiated pile slugs. These gases are converted successfully to solid waste-disposal units by collecting them quantitatively in evacuated refrigerated bombs filled with activated carbon. These bombs are closed by means of valves, disconnected from the system, and incorporated in concrete for sea burial.

D. Types of Waste Packages Produced at UCRL, Berkeley

1. Steel drums, dry waste

These are used 55-gallon oil or fuel drums from various sources. One of the ends is cut out and 6 inches of concrete is placed in the bottom of the drum. A length of used 3/8-inch steel cable is placed in the concrete before it hardens. The cable is of sufficient length to form a loop above the top of the drum, for handling of the drum after it has been filled. The prepared drums are filled with sacks of dry waste layered with concrete. During the concreting the concrete is tamped so that voids in the drum are filled as much as possible. About 3.5 ft^3 of dry waste material is disposed of in each 55-gallon drum. A concreted drum must weigh at least 550 lb to sink at sea. The finished drums are stored on their sides in the waste-disposal area until shipment. (Storing the drums on their sides prevents seepage of rain water into possible voids around the rim of the drum. A considerable amount of monitoring work is saved in this way, since all liquid that runs out of a drum must be assayed for radioactivity to ascertain if a container failure has occurred.)

2. Standard concrete blocks

These blocks (approximately 4 × 6 × 7 ft) are made up from time to time for the disposal of gloved boxes and other bulky equipment. A knock-down form with the exception of one side is built by the carpenters on a wood pallet. The bottom of the block is poured first and permitted to set. The equipment to be disposed of is placed on the concrete bottom in such a way that none of the waste touches the form. The form is then completed by the carpenters and concrete is poured to completely fill it.

In some cases--for example, in the disposal of an enclosure used in a 6-inch cave operation--the air inside the box is slowly expelled through a filter as it is displaced by the concrete actually poured into the box. The concrete pouring is continued until the form and box are completely filled.

3. Special concrete blocks

These blocks are designed to meet specific needs in the disposal of radioactive waste. They vary in size and shape and may provide shielding of up to 18 inches of concrete. Some examples are:

a. Concrete block for disposal of solid-waste containers from a 6-inch lead cave rim.

b. Block for disposal of high-level liquid waste. This type of block is usually of the standard 4 × 6 × 7-ft dimensions, but differs from the regular block in having an empty 55-gallon drum centered in the form. The block is then poured so that the drum is completely surrounded by concrete except for three pipes through the lid of the 55-gallon drum. Liquid waste is pumped into the drum and solidified with cement-vermiculite. After solidification, the block is capped with cement and stored until shipment.

4. Steel drums, liquid waste

A package of this type consists of a 30-gallon drum with a special lid which is set into a 55-gallon drum (see II D 1). After solidification is complete the space between the 55-gallon drum and the 30-gallon drum is filled with concrete. The drum, capped with concrete, is then ready for disposal.

E. Average Yearly Amounts of Waste Handled at UCRL, Berkeley

In spite of the fact that large quantities of radioactive materials are handled at UCRL, Berkeley (at one time 10% of the radioisotopes produced at Oak Ridge were being processed at UCRL), the yearly average volume of radioactive wastes produced is quite low. Average volumes for the last three years are as follows.

1. Solid wastes (estimated unprocessed dry volume)

55-Gallon drums	3500 ft^3
Standard blocks	350 ft^3
Special blocks	25 ft^3

2. Liquid wastes (estimated liquid volume before solidification)

Low-level wastes	950 gallons
High-level wastes	30 gallons
Organic wastes	25 gallons

F. Cost of Waste Disposal

A survey made in 1957 established the following cost figures. The figures are estimates based on material and labor costs at that time.

1. Steel drums, dry waste (per drum)

Health Chemistry labor	$ 5	
Other labor	3	
Materials	5	
Miscellaneous (transporation, etc.)	2	
Sea-disposal charge	4	
Cost per drum of dry waste	$ 19	
Cost per ft^3 (3.5 ft^3/drum		$6

2. Standard UCRL concrete Blocks (per block)

Health Chemistry labor	$ 50	
Other labor	140	
Material	140	
Equipment and miscellaneous (transportation, etc.)	15	
Sea-disposal charge	35	
Cost per block	$380	
Cost per ft^3		$9.50

3. Steel drums, liquid waste (per drum)

Health Chemistry labor	$ 4	
Other labor	1	
Materials (special drums, modification, etc.)	20	
Cement-vermiculite	3	
Miscellaneous (transportation, etc.)	1	
Sea disposal charge	4	
Cost per drum	$ 33	
Cost per gallon (average 17.5 g/drum)		$2

The estimate of $2 per gallon of solidified liquid waste may seem high until we consider it in the over-all picture of the cost of waste disposal. Actually it represents a considerable amount of money and time saved. The reason for this is as follows.

About 1000 gallons of potentially radioactive liquid is produced daily the chemistry laboratories at UCRL, Berkeley. That means that if the "concentrate and confine" procedures were not followed there would be an estimated 1000 gallons of active liquids to dispose of. However, the chemists and technicians working with radioactive isotopes cooperate by using techniques and facilities made available to them by Health Chemistry, and it is possible to keep the radioactive liquid waste to a small volume (about 5 gallons per day) and to segregate it from the nonactive laboratory wastes discharged through the "Acid Waste System." (See Sect. II B 5.) Therefore, instead of decontaminating or disposing of 1000 gallons of liquid waste per day, the Health Chemistry technicians need only solidify 5 gallons of active waste and assay samples from 995 gallons of other liquid to establish that no accidental contamination has occurred.

The adjusted cost for liquid waste disposal at UCRL, Berkeley, based on a potential of 1000 gallons per day, becomes about 1.4 cents per gallon. For example:

Cost to solidify 5 gallons of active waste	$10	
Cost of operation and sampling of "Acid Waste System" per day	4	
Total	$14	
Cost per potential gallon (based on 1000-gallon estimate)		$0.014

III. Conclusions

A system for the safe packaging of laboratory hot wastes for storage or sea disposal has been described. The system is based on quantity control and separation of the radioactive wastes from non-radioactive wastes during laboratory work. In this way economy is achieved and volumes can be kept to reasonable levels to permit packaging in waste containers of such integrity for prolonged surface storage or proper handling for sea disposal. The methods described have been applied safely to laboratory wastes for several years. Extension of these methods to large-volume wastes as encountered in production facilities would require further study.

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COMPARATIVE COSTS OF SEA DISPOSAL AND LAND BURIAL FOR THE RADIOACTIVE WASTES OF THE LAWRENCE RADIATION LABORATORY

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ABSTRACT

A study has been made of comparative costs of disposal of radioactive wastes at sea and by burial, taking into account such factors as loading, storage, and transportation by various means.

Introduction

This report is a comparative cost study of radioactive waste disposal for the Lawrence Radiation Laboratory (UCRL). In particular, it compares the costs of sea disposal in depths of 1000 fathoms and of 2000 fathoms off the California coast with land burial of the wastes at the Hanford Atomic Products Operation (HAPO), Richland, Washington, at the National Reactor Testing Station (NRTS), Idaho Falls, Idaho, or at the Nevada Test Site (NTS), Nevada. In the comparison, the cost of utilizing a commercial waste-disposal firm is also shown as well as a tentative cost of using the Military Sea Transport Service (MSTS). It is understood that the commercial firm would utilize sea disposal.

In comparing sea disposal versus land disposal, it is assumed that all shipments start with a concreted 55-gallon drum containing the radioactive wastes. The purpose of the concrete is to encase and keep under safe control the radioactive materials within. With respect to sea disposal it has the additional attributes of providing the necessary weight for sinking at sea and of inhibiting, over long periods of time, the diffusion of the radioactive contents into the surrounding environment. It is assumed that this latter attribute is also desired in land burial. It should be recognized and understood that the 55-gallon drum serves only as a concrete pouring form and may be replaced by more economical forms as they are developed.

Lawrence Radiation Laboratory wastes consist occasionally of packages other than the concreted 55-gallon drum. However, for simplicity of comparison all costs have been reduced to the cost for a concreted 55-gallon drum having an average weight of 750 pounds.

To reduce the cost per 55-gallon container to costs per cubic foot of radioactive wastes or per gallon of liquid waste in the container, one can assume that about 4 cubic feet of uncompacted dry waste packages or 20 gallons of liquid waste are placed in a UCRL concreted drum. With the dry waste the added concrete fills the voids around and between the waste packages and serves to compact the wastes to some degree.

It is assumed in the survey that the concreted 55-gallon drum will have the ICC approval necessary to obviate any special Health Physics escort or services during transportation and handling. A brief examination shows that such services, if required, would add about 75 cents per drum to the cost of sea disposal. Most of this cost would arise during truck transportation. With land burial such services would add from $2 to $3 per drum to the cost of truck transportation to the site, and would be almost impossible for rail shipments unless arrangements for special and expeditious shipment were made.

All cost estimates have an inherent inaccuracy in that commercial costs were not based on firm bids but on estimates furnished by such concerns. In some cases there is no applicable classification for which a transportation rate has been determined. Thus transportation rates have been estimated from rates previously established for items somewhat similar in size or shape.

The best transportation rates are based on at least a given minimum quantity per shipment. For truck and rail shipments, this produces no problem, since full truckloads or railcar loads can easily be made without straining UCRL's storage capacity. For sea disposal, however, optimum rates might require making disposal trips about once or twice a year. This creates no problem so far as Livermore is concerned. However, Berkeley's

storage capacity is limited to 6 to 8 weeks' accumulation. In such cases consideration has been given to transportation from Berkeley to Livermore for interim storage pending a sea trip. Another possibility is interim storage at a commercial facility, which is quite close to a dock near the Oakland, California airport.

It is recognized that installations burying radioactive wastes within their own sites have no containment of such wastes except the plastic bags or cardboard cartons in which the wastes are collected. Other facilities shipping to these installations have their wastes buried in shipping containers such as sealed drums or specially fabricated metal boxes that can be tightly sealed.

Although it was not considered the purpose of this survey to investigate other containers for UCRL's radioactive wastes, general comments might be made. The cost of special fabricated metal containers or of sealable 55-gallon drums can amount to slightly more than $1.00 per cubic foot of packaged wastes. The cost of materials for a 55-gallon concreted drum amounts to about $1.00 per cubic foot of packaged wastes.

The major portion of disposal costs arises from loading and unloading operations and from rail and truck transportation between the point of origin and the disposal area. While such costs are generally quoted in terms of cost per ton handled or cost per hundred pounds transported for a given minimum loading, the basic factor is the number of packages handled and the volume occupied per package. For packages with a lower density and therefore smaller possible truckloads or carloads the transportation rates are higher per unit weight.

Savings that could be effected in some cases by palletizing loads, changing package sizes, loading barges more efficiently, or providing greater storage space in Berkeley were not examined.

Summary

The average cost estimate for commercial sea disposal at 1000 fathoms of UCRL's radioactive waste from Berkeley and Livermore ranged from $7.48 per drum to $8.00 per drum, assuming utilization of one of the two commercial tug and barge concern investigated. Between these two limits, prices were determined by the number of drums per sea trip and by whether Livermore or a San Leandro storage yard were used for interim storage of Berkeley's drums. Because of the need for extra transportation and storage of Berkeley drums, it is not axiomatic that the greatest number of drums per sea trip results in the lowest cost. Costs were computed for 500, 1000, and 3000 drums per sea trip.

With the second barge and tug concern, the average cost for sea disposal at 1000 fathoms ranged from $8.53 per drum for 1850 drums per sea trip to $20.47 per drum for a 500-drum sea trip. It is the author's belief that the latter figure is unrealistic (see discussion under Sea Disposal Section) and should probably be more like $10 per drum for a 500-drum sea trip.

The extra cost for disposal at 2000 fathoms instead of 1000 fathoms ranged from $0.33 per drum for a 3000-drum sea trip to $2.00 per drum for a 500-drum sea trip, for either barge and tug concern referred to above.

The cost for disposal utilizing a commercial waste-disposal concern[1] has been given as $14.50 per 55-gallon drum. The tentative cost for disposal via MSTS is $5.28 per drum.

The cost estimate for land burial at HAPO, NRTS, and NTS ranged from $14.05 to $15.89 per drum if rail transportation were used, and from $14.90 to $19.25 per drum if through truck transportation were used.

Sea Disposal

As stated in the introduction, costs of sea disposal were examined, taking into consideration various numbers of drums per sea trip. For quantities over 500 drums per sea trip provisions would be needed for interim storage of drums in excess of Berkeley's upper storage limit of approximately 200 drums. Consideration has been given to interim storage on the Livermore Site and at a commercial facility located in San Leandro, California close to a dock at the Oakland Airport.

Sea disposal costs were determined for dumping at 1000 and at 2000 fathoms. This involves barge trips of 1 and 2 days, respectively. The extra cost for 2000-fathom dumping amounts to $1000 per trip.

In the investigation of available sea-going barging concerns in the bay area, two were found, one of which was interested only in the barge rental and sea towing aspects, while the other could handle all arrangements after disposition of UCRL's drums at a dock, including arrangements for a dock, and loading of the barge.

Docks considered were those in the East Bay Area. There are several docks in Oakland fronting on San Francisco Bay in or near the Estuary separating Oakland and Alameda. In addition there is a dock near the Oakland Airport. Any of these docks would be almost equally convenient to the Berkeley and Livermore laboratories. Docks in Richmond were not investigated but could probably be used. These would be more convenient to Berkeley than to Livermore.

Summation of the cost estimates for various combinations of the number of drums per sea trip, storage or dock facilities, tug and barge concerns, and depth of dumping are given in the Appendix, Cases A through E.

The factors that enter into sea disposal include some of the following:

1. Loading or unloading of a truck at UCRL by UCRL.
2. Transportation by commercial truck:
   a. Berkeley to dock; unloading on dock.
   b. Livermore to dock; unloading on dock.
   c. Berkeley to Livermore for interim storage.
   d. Berkeley to a commercial facility; interim storage
   e. Commercial facility to dock; unloading on dock.
3. Dock rental.

[1] Proposal of Nuclear Engineering Co., Inc., March 25, 1957.

4. Barge rental while docked.

5. Barge loading.

6. Tug and barge charges for sea trip and dumping.

1. Truck Loading or Unloading at UCRL by UCRL

Truck loading at UCRL of radioactive waste drums has been carried on for the last 10 years. The cost of loading is about 75 cents per drum. The cost of unloading at Livermore for interim storage has been estimated as the same.

2. Transportation by Commercial Truck

Costs of transportation by commercial truck have been estimated[2] for the various hauls that are considered in this survey.

2a. Transportation from UCRL-Berkeley to Dock; Unloading at Dock

It has been estimated that truck transportation from UCRL-Berkeley to an Oakland dock and unloading at the dock would be $0.25 per hundred pounds for a 40,000-pound minimum truck load. This amounts to $1.88 for a 750-pound drum, with a 53-drum minimum truck load. The trucks to be used have space for 64 drums. Weight limitations would probably keep a load to about 55 drums. The cost of transport only (no unloading) to a Port Chicago dock has been estimated at $1.88 per drum.

2b. Transportation from UCRL-Livermore to Dock; Unloading at Dock

The cost of transportation from UCRL-Livermore to an Oakland dock and unloading has been estimated to be $2.25 for the 750-pound drum with the same 53-drum minimum truck load. The cost of transportation only (no unloading) to a Port Chicago dock has been estimated at $1.88 per drum.

2c. Transportation from UCRL-Berkeley to UCRL-Livermore; Loading and Unloading by UCRL

The cost of transporting drums from UCRL-Berkeley to UCRL-Livermore has been estimated at $1.88 per 750-pound drum, with a 53-drum minimum truck load. UCRL costs of both loading and unloading the truck would be in addition to these.

2d. Transportation from UCRL-Berkeley to Commercial Facility; Interim Storage

The cost of transportation from Berkeley to a storage yard in San Leandro, California has been estimated at $1.50 per drum, with the same 53-drum minimum. Storage charges would be about 3 cents per square foot per month or about 12 cents per drum per month.

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2 Bigge Drayage Company, San Leandro, California, Mr. Doty and Mr. Brooks.

2e. Transportation from Commercial Facility to Dock; Unloading at Dock

The cost of transportation from the storage yard in San Leandro to Oakland Airport dock plus and unloading, together with the furnishing of a crane for barge loading, has been estimated at $2.18 per drum.

3. Rental of Dock Space

The dock charges used in this survey are those obtained from Howard Terminals,[3] who are located on the Oakland Estuary and who have a rail crane capable of handling up to 7.5 tons. The charges comprise a $1.42-per-ton service charge plus a $0.35-per-ton wharfage fee, making a total of $1.77 per ton. This amounts to $0.66 per 750-pound drum. One is allowed 5 days in which to make a loading.

4. Barge Rental While Docked

This was treated by one barge and tug operator[4] as a separate item, and amounted to about $75 per day. Assuming that 500 drums can be loaded in one day, one has a charge of $0.15 per drum.

5. Barge Loading

An estimate of the cost of barge loading has been obtained from the California Stevedore and Ballast Co.[5] It was estimated that 75 to 80 drums could be loaded per hour. Stevedore gang and crane charge would amount to $3.50 per ton handled. In addition there would be a service or clerical charge of $2.25 to $2.50 per ton. Taking a total of $6.00 per ton, one obtains a cost of $2.25 per drum. There is uncertainty whether or not stevedores might require a penalty payment because of the type of cargo handled.

6. Tug and Barge Charges for Sea Trip and Dumping

One barge and tug organization[4] in the Bay Area is acquiring a barge for the handling and dumping of sand, mud, or gravel. It has a hold shaped like a half cylinder with a radius of 15 feet and a length of 130 feet. The cargo is dumped by sinking one end of the barge and letting the cargo slide out. The load capacity is 1125 tons. In terms of UCRL's 750-pound drum, such a barge could handle about 3000 drums. Space limitations give about the same load limit.

[3] Howard Terminals, Oakland, California, Mr. Jack Bentry

[4] Seaborn Towing Service, Mr. Seaborn

[5] California Stevedore and Ballast Co., San Francisco, California, Mr. Ebey

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The estimated cost for use of the barge and tug would be $1000 per 24-hour day. Dumping at 1000 fathoms would take 1 day, while dumping at 2000 fathoms would require 2 days. At the maximum capacity of 3000 drums, the cost of the sea trip would be $0.33 per drum for a 1000-fathom dump and $0.66 per drum for a 2000-fathom dump. For a load of 1000 drums, the costs would be $1.00 and $2.00 per drum respectively, while for a 500-drum load the costs would be $2.00 and $4.00 per drum respectively for the 1000-fathom and 2000-fathom dumps.

This particular organization does only barge hauling. All other steps would have to be separately negotiated.

3, 4, 5 and 6 combined(commercial barging):
All Operations and Costs from Loading of Barge in Oakland to Dumping at Sea

Another barge and tug organization[6] has a sea-going barge located in the Seattle, Washington area and is acquiring another for use in the San Francisco Bay Area. This would be a flat-deck barge measuring 120 x 44 feet and having a load capacity of 700 tons. The barge is dumped by sinking one end and allowing the load to slide off. A single layer of drums standing upright on the deck in an area 100 feet by 40 feet would be 1000 drums or a load of about 375 tons. It is understood, however, that a barge would be loaded by stacking the drums on their sides in order to obtain a high center of gravity and make the load dumping easier. The 700-ton limitation permits a load of about 1850 drums.

The estimated cost for a trip, including all operations from loading the barge at an Oakland dock to dumping, is $9,000 for a 1000-fathom dump and $10,000 for a 2000-fathom dump. Although it was stated that it might be possible to share the costs of a sea dump (for a less-than-full-load drum shipment) with some other commercial dumping operation, the costs have been derived as if the concreted drums were the only cargo.

For a maximum load of 1850 drums, the cost comes to $4.87 per drum for a 1000-fathom dump and $5.41 per drum for a 2000-fathom dump. For a load of 1000 drums, the costs are $9.00 and $10.00 per drum respectively, while for a 500-drum load the costs are $18.00 and $20.00 per drum for the 1000-fathom and 2000-fathom dumps.

It is believed that the values per drum given above for a 1000- and a 500-drum load are not realistic. The difference between the total cost for a 1000-fathom dump and a 2000-fathom dump is $1000, which as a charge for one day's rental of a barge and tug is comparable to other estimates obtained. Assuming that the 1000-fathom dumping is a 1-day sea trip, one has a charge of $8000 for dock rental, barge rental during loading, and barge loading. For an 1850-drum load this amounts to $5.32 per drum for these charges, which can be compared with the $3.06 per drum that results from the total of the individual estimates for these factors.

However, the total charge for dock rental, barge rental during loading, and barge loading ought to be in proportion to the number of drums handled and not a fixed charge regardless of the number of drums handled; i.e., the cost per drum should remain approximately the same.

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[6] Bay Cities Transportation, Mr. Escher.

3, 4, 5, and 6 combined (MSTS):
All Operations from Truck Unloading at Concord or Port Chicago to Dumping at Sea

The Military Sea Transport Service (MSTS) has indicated[7] its ability to handle the sea disposal of radioactive waste from the Concord Naval Ammunition Depot. Not much is yet known about the cost of this service, but it has been indicated[7,8] that the charge for these operations would be $10 per measured ton (40 cubic feet). Since a 55-gallon drum occupies a cubical space of about 12 cubic feet, one obtains a cost for these services of $3.00 per drum.

Land Burial

For land burial, the costs were examined for three possible burial areas: (a) within the Hanford Atomic Products Operation (HAPO) at Richland, Washington, (b) within the National Reactor Testing Station (NRTS) at Idaho Falls, Idaho, and (c) within the Nevada Test Site (NTS) in Nevada. No estimates were obtained directly from any of these possible burial sites on local transportation costs, loading and unloading costs, or burial costs. Instead, estimates were made by extending local estimates or examining data available in a cost survey on radioactive waste disposal practices in the atomic energy industry.[9]

With respect to rail transportation, the nearest commercial railhead to the Hanford Atomic Products Operation is Hanford Works,[10] from which carload deliveries are interchanged to a government railroad for delivery within the facility. How close the government railraod is to a possible burial site is not known, but it is assumed to be less than 10 miles.

The nearest commercial railhead to the National Reactor Testing Station is Scoville, Idaho.[10] It is reported that[11] there is government railroad for making carload deliveries within the facility, and it is again assumed that the government railroad comes within 10 miles of a possible burial site.

The nearest commercial railhead to the Nevada Test Site is in Las Vegas, Nevada.[10] There is no public crane in Las Vegas, but it is understood[11] that a private crane is available for $15 per hour. The distance from Las Vegas to Camp Mercury is about 65 miles and it is assumed that the distance to a possible burial site is another 35 miles for a total trucking of 100 miles.

With respect to commercial trucking, it is assumed that such trucks will be able to go directly to the burial sites.

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7 Memo, Robert Hughey, AEC-SAN to E. C. Shute, AEC-SAN, Dec. 15, 1958.

8 Personal communication, Robert Hughey to E. Nielsen.

9 Arnold B. Joseph, Radioactive Waste Disposal Practices in the Atomic Energy Industry--A Survey of the Costs, NYO-7830, Dec. 1955.

10 Transportation Director, AEC Manual Chapter 5205.

11 Southern Pacific Company, Oakland, California, Mr. Donald Wicks and Mr. Cliff D. Varga.

Summations of the cost estimates for land burial utilizing truck and rail transportation to the burial facility are given in the Appendix, Cases F and G.

The factors that would enter into land burial of UCRL's radioactive wastes are the following:

1. Loading of truck at UCRL by UCRL.
2. Transportation by commercial truck.
   a. to Oakland, California rail terminal, or
   b. directly to HAPO, NRTS, or NTS.
3. Loading of railcar in Oakland, California, and shoring of load.
4. Rail transportation to HAPO, NRTS, or NTS.
5. Unloading of railcar at railhead.
6. Transportation from railhead to burial site.
7. Unloading at burial site.
8. Direct burial costs: excavation, and cover of burial site.
9. Guarding and checking of burial site.

1. Loading of Truck at UCRL by UCRL

The cost would be same as for sea disposal operations; i.e., $0.75 per drum.

2a, 3. Transportation by Truck to Oakland, California Rail Terminal

Transportation by commercial truck from UCRL to the Southern Pacific Railroad freight depot in Oakland, California and loading of the railcar has been estimated[2] to be $0.20 per hundred pounds for a Berkeley load and $0.25 per hundred pounds for a Livermore load, with 40,000-pound minimum load. This amounts to $1.50 per drum from Berkeley and $1.88 per drum from Livermore. The freight depot has a public crane that can handle up to 50 tons.

It has been estimated that the cost of shoring a load in a railcar is about $100 per railcar for labor and materials. The particular type of railcar recommended[11] is the gondola car, having a bed 51 feet 6 inches long and 9 feet 2 inches wide, and an upper load limit of about 50 tons. Such a car can carry 125 drums for a total weight slightly under 47 tons. The cost of shoring would thus amount to $0.83 per drum.

The total cost of transporting to the freight depot, loading the railcar, and shoring amounts to $2.33 per drum for Berkeley drums and $2.71 per drum for Livermore drums.

2b. Truck Transportation to HAPO, NRTS, or NTS

The costs of transportation by truck from UCRL directly to HAPO,[12] to NRTS,[12] or to NTS[13] have been estimated as follows:

	Minimum load per truck		Cost	
To	lbs.	Drums	per 100 lb.	per drum
HAPO	24,000	32	$2.20	$16.50
NRTS	35,000	47	2.18	16.35
NTS	24,000	32	1.50	11.25

4. Rail Transportation to HAPO, NRTS, or NTS

The costs of rail transportation from the Southern Pacific Company freight depot in Oakland, California to the nearest railhead at HAPO, NRTS, and NTS have been estimated[11] from charges for somewhat similar items(metal-reinforced concrete slabs; molded cement blocks). The locations of the railheads, the minimum car loadings, and transportation costs for these items are as follows:

Estimated costs per drum of concreted wastes, based on comparable materials

		Railhead		
		HAPO Richland, Wash.	NRTS Scoville, Utah	NTS Las Vegas, Nev.
Metal reinforced concrete slabs; minimum loading, 60,000 lb	per 100 lb	$0.89	$0.92	$0.73
	per 750 lb	6.68	6.90	5.48
Molded cement blocks; minimum loading, 50,000 lb	per 100 lb	1.02	1.04	0.85
	per 750 lb	7.65	7.80	6.38
Average	per drum	7.17	7.35	5.93

The cost of shuttling a carload from the commercial railhead into the facility is considered to be negligible and at least within the limits of uncertainty of other burial-site costs.

[12] Garrett Freightlines, Inc.

[13] Wells Cargo, Inc.

5. Unloading of Railcar at Railhead

An estimate for the cost of this operation was taken as an average of UCRL cost for the loading of a truck and the estimated cost of loading a barge. This amounts to $1.50 per drum. Demurrage charges, if incurred, are considered to be within the limits of uncertainty of this cost.

6. Truck Transportation from Railhead to Burial Site

At NTS the transportation cost is estimated at $0.40 per hundred pounds or $3.00 per drum. At HAPO and NRTS, this is estimated at $0.30 per drum.

7. Unloading at Burial Site

The cost of unloading at the burial site is assumed to be $1.00 per drum.

8. Direct Burial Costs: Excavation, and Cover of Burial Site

In a previous cost-survey report[9] it is stated that at the National Reactor Testing Station the cost of burial is $0.36 per cubic foot of waste. For a 55-gallon drum of approximately 7 cubic feet, this would be $2.52 per drum. However, it is noted that charges made by the National Reactor Station to Rocky Flats for burial of wastes amounts to about $0.10 per cubic foot, which would be equivalent to $0.70 per 55-gallon drum. These charges to Rocky Flats must have also included truck-unloading costs.

At the Oak Ridge National Laboratory the cost-survey report[9] shows a cost of $0.22 per cubic foot for land burial operations. This would amount to $1.54 per 55-gallon drum. Charges made to other installations for burial of their wastes were about $0.17 per cubic foot, and the charge presumably included hauling 9 miles from a railhead to the dump site.

The waste-disposal service at the Oak Ridge National Laboratory available to industrial and scientific organizations is described in the 1957 edition of their Isotopes catalog. There it is stated that their disposal price for a package weighing less than 500 pounds and measuring less than 4 by 4 by 4 feet is $10.00, and that the price for heavier or larger packages is supplied upon request. For the maximum-size package of 64 cubic feet this charge amounts to 16 cents per cubic foot. For a 1-cubic-foot package this would be $10.00 per cubic foot, and for the 55-gallon drum $1.43 per cubic foot. These figures imply that the major cost is in the package handling from railhead to burial ground, and that burial cost is of the order of 15 cents per cubic foot or less.

From all of the above, a value of $1.00 per drum is taken as the burial cost.

9. Guarding and Checking of Burial Site

It is assumed that one would want to assure in perpetuity that burial sites will not be re-excavated. If burial sites are located within government-owned areas, serving other functions as well, one can be reasonably well assured that there will be adequate guarding for as long as the government exists. It is assumed that such protection, functioning as a supplementary operation, is a negligible cost per item buried.

Despite what logic may say regarding any public danger, the assuaging of public anxiety will probably require periodic sampling and testing of underground waters, lakes, and rivers in the vicinity of burial sites. Since this is a repetitive operation over many, many years and since the resultant cost per item buried is dependent upon the size of the burial area, no attempt was made to estimate the cost.

APPENDIX

Compilation of cost estimates for various assumed ways in which sea disposal or land disposal might be effected.

Sea Disposal

		Page
Case A-1	3000 drums per sea trip	14
Case A-2		15
Case B	1850 drums per sea trip	16
Case C-1		17
Case C-2	1000 drums per sea trip	18
Case C-3		19
Case D-1	500 drums per sea trip	20
Case D-2		21
Case E	MSTS (Military Sea Transport Service)	21

Land Burial

Case F	Rail transportation to burial facility	22
Case G	Truck transportation to burial facility	22

Case A-1 Dumping at 1000 fathoms; 3000 drums per sea trip; 200 drums directly from Berkeley to Oakland dock; 1300 drums from Berkeley to Livermore to Oakland dock; 1500 drums from Livermore to Oakland dock; Seaborn Towing Service as the barge and tug organization. (Costs for 2000-fathom dumping, if different, shown in brackets)

Item	Cost per drum			
	Berkeley to dock	Berkeley to Livermore to dock	Livermore to dock	Average
1. UCRL truck loading and unloading	$0.75	$2.25	$0.75	
2. Transportation				
a. Berkeley to dock	1.88			
b. Livermore to dock		2.25	2.25	
c. Berkeley to Livermore		1.88		
3. Dock rental	0.66	0.66	0.66	
4. Barge rental	0.15	0.15	0.15	
5. Barge loading	2.25	2.25	2.25	
Subtotal	$5.69	$9.44	$6.06	$7.50
6. Sea trip	0.33 [0.66]	0.33 [0.66]	0.33 [0.66]	
Total	$6.02 [6.35]	$9.77 [10.10]	$6.39 [6.72]	$7.83 [8.16]

Case A-2 Dumping at 1000 fathoms; 3000 drums per sea trip; 1500 drums from Berkeley to commercial facility in San Leandro for storage, 9 months average storage time, thence from San Leandro storage to Oakland airport dock; 1500 drums from Livermore to Oakland airport dock; Seaborn Towing Service as the barge and tug organization. (Costs for 2000-fathom dumping, if different, shown in brackets)

Item	Cost per drum		
	Berkeley to San Leandro to dock	Livermore to dock	Average
1. UCRL truck loading	$0.75	$0.75	
2. Transportation and storage			
c. Livermore to dock		2.25	
d. Berkeley to San Leandro storage	2.58		
e. San Leandro to dock	2.18		
3. Dock rental	0.66	0.66	
4. Barge rental	0.15	0.15	
5. Barge loading	2.25	2.25	
Subtotal	$8.57	$6.06	$7.32
6. Sea trip	0.33 [0.66]	0.33 [0.66]	
Total	$8.90 [9.23]	$6.39 [6.72]	$7.65[7.98]

Case B Dumping at 1000 fathoms; 1850 drums per sea trip; 200 drums directly from Berkeley to an Oakland dock; 725 drums from Berkeley to Livermore to Oakland dock; 925 drums from Livermore to Oakland dock; Bay Cities Transportation as the barge and tug organization. (Costs for 2000-fathom dumping, if different, shown in brackets)

Item	Cost per drum			
	Berkeley to dock	Berkeley to Livermore to dock	Livermore to dock	Average
1. UCRL truck loading and unloading	$0.75	$2.25	$0.75	
2. Transportation				
a. Berkeley to dock	1.88			
b. Livermore to dock		2.25	2.25	
c. Berkeley to Livermore		1.88		
Subtotal	$2.63	$6.38	$3.00	$4.28
3. Dock rental 4. Barge rental 5. Barge loading 6. Sea trip	4.87 [5.41]	4.87 [5.41]	4.87 [5.41]	
Total	$7.50 [8.04]	$11.25 [11.79]	$7.87 [8.41]	$9.15[9.69]

Case C-1 Dumping at 1000 fathoms; 1000 drums per sea trip; 200 drums directly from Berkeley to Oakland dock; 300 drums from Berkeley to Livermore to Oakland dock; 500 drums from Livermore to Oakland dock; Seaborn Towing Service as the barge and tug organization. (Costs for dumping, if different, in 2000 fathoms shown in brackets)

Item	Cost per drum			
	Berkeley to dock	Berkeley to Livermore to dock	Livermore to dock	Average
1. UCRL truck loading and unloading	$0.75	$2.25	$0.75	
2. Transportation				
a. Berkeley to dock	1.88			
b. Livermore to dock		2.25	2.25	
c. Berkeley to Livermore		1.88		
3. Dock rental	0.66	0.66	0.66	
4. Barge rental	0.15	0.15	0.15	
5. Barge loading	2.25	2.25	2.25	
Subtotal	$5.69	$9.44	$6.06	$7.00
6. Sea trip	1.00 [2.00]	1.00 [2.00]	1.00 [2.00]	
Total	$6.69 [7.69]	$10.44[11.44]	$7.06 [8.06]	$8.00[9.00]

Case C-2 Same as Case C-1, except using Bay Cities Transportation as the barge and tug organization.

Item	Cost per drum			
	Berkeley to dock	Berkeley to Livermore to dock	Livermore to dock	Average
1. UCRL truck loading and unloading	$0.75	$2.25	$0.75	
2. Transportation				
a. Berkeley to dock	1.88			
b. Livermore to dock		2.25	2.25	
c. Berkeley to Livermore		1.88		
Subtotal	$2.63	$6.38	$3.00	$3.94
3. Dock rental 4. Barge rental 5. Barge loading 6. Sea trip	9.00 [10.00]	9.00 [10.00]	9.00 [10.00]	
Total	$11.63 [12.63]	$15.38 [16.38]	$12.00 [13.00]	$12.94 [13.94]

Case C-3 Same as C-1, except storage of the 300 Berkeley drums is made at a San Leandro commercial facility, the dock used is the one near Oakland airport, and the average storage period is 4 months.

Item	Cost per drum			
	Berkeley to dock	Berkeley to San Leandro to dock	Livermore to dock	Average
1. UCRL truck loading	$0.75	$0.75	$0.75	
2. Transportation and storage				
a. Berkeley to dock	1.50			
b. Livermore to dock			2.25	
c. Berkeley to San Leandro storage		1.98		
e. San Leandro to dock		2.18		
3. Dock rental	0.66	0.66	0.66	
4. Barge rental	0.15	0.15	0.15	
5. Barge loading	2.25	2.25	2.25	
Subtotal	$5.31	$7.97	$6.06	$6.48
6. Sea trip	1.00 [2.00]	1.00 [2.00]	1.00 [2.00]	
Total	$6.31 [7.31]	$8.97 [9.97]	$7.06 [8.06]	$7.48 [8.48]

Case D-1 Dumping at 1000 fathoms; 500 drums per sea trip; 250 drums from Berkeley to Oakland dock; 250 drums from Livermore to Oakland dock; Seaborn Towing Service as the barge and tug organization. (Costs for 2000-fathom dumping, if different, shown in brackets.)

Item	Cost per drum: Berkeley to dock	Cost per drum: Livermore to dock	Average
1. UCRL truck loading	$0.75	$0.75	
2. Transportation			
a. Berkeley to dock	1.88		
b. Livermore to dock		2.25	
3. Dock rental	0.66	0.66	
4. Barge rental	0.15	0.15	
5. Barge loading	2.25	2.25	
Subtotal	$5.69	$6.06	5.88
6. Sea trip	2.00 [4.00]	2.00 [4.00]	
Total	$7.69 [9.69]	$8.06 [10.06]	$7.88 [9.88]

Case D-2 Same as D-1, except utilizing Bay Cities Transportation as the barge and tug organization.

Item	Cost per drum: Berkeley to dock	Livermore to dock	Average
1. UCRL truck loading	$0.75	$0.75	
2. Transportation			
a. Berkeley to dock	1.88		
b. Livermore to dock		2.25	
Subtotal	$2.63	$3.00	2.82
3. Dock rental 4. Barge loading 5. Barge rental 6. Sea trip	18.00 [20.00]	18.00 [20.00]	
Total	$20.63 [22.63]	$21.00 [23.00]	$20.82 [22.82]

Case E Sea disposal through the Military Sea Transport Service; trucking from Berkeley and Livermore to storage area at Concord Naval Ammunition Depot or to Port Chicago dock.

1. UCRL truck loading	$0.75
2. Transportation	1.88
Subtotal	$2.63
MSTS charges*	3.00*
Total	$5.63

* There is some question as to the amount of these charges.

Case F Trucking from UCRL to Oakland, California rail freight depot; rail transportation to burial installation railhead. (Costs are shown for Berkeley; corresponding costs for Livermore are given in brackets.)

Item	Cost per drum HAPO	NRTS	NTS
1. UCRL truck loading	$0.75	$0.75	$0.75
2a, 3. Transportation to Oakland freight depot, railcar loading, shoring	2.33 [2.71]	2.33 [2.71]	2.33 [2.71]
4. Rail to railhead	7.17	7.35	5.93
5. Unloading railcar	1.50	1.50	1.50
6. Transportation to burial site	0.30	0.30	3.00
7. Unloading at burial site	1.00	1.00	1.00
8. Burial	1.00	1.00	1.00
Total	$14.05 [14.43]	$14.23 [14.61]	$15.51 [15.89]
Average, Berkeley, Livermore	14.24	14.42	15.70

Case G Trucking from UCRL at Berkeley or Livermore directly to burial site.

Item	Cost per drum HAPO	NRTS	NTS
1. UCRL truck loading	$0.75	$0.75	$0.75
2b. Transportation to burial site	16.50	16.35	11.25
7. Unloading at burial site	1.00	1.00	1.00
8. Burial	1.00	1.00	1.00
Total	$19.25	$19.10	$14.90

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OFFICE FOR ARMY REACTORS, DIVISION OF REACTOR DEVELOPMENT, AEC

Abstract

The APPR-1 located at Fort Belvoir, Virginia has operated successfully for 20 months. An environmental monitoring program has been in effect since the pre-construction period to insure that there is no increase in radioactivity in the area of the APPR-1. Effluent monitors provide a control for the discharge of gaseous wastes to the atmosphere and liquid wastes to the Potomac River. All gaseous wastes to date have been vented to the atmosphere. Liquids, the greatest contributor to the radioactive wastes at the APPR-1, have been collected and discharged safely to the river. Solid wastes are accumulated irregularly and are disposed of by approved methods. In general, the management of radioactive wastes at the APPR-1 is a simple operation.

Waste Management at the APPR-1

The APPR-1 is a 10 megawatt pressurized water reactor located adjacent to Gunston Cove in the confines of the Engineer Research and Development Laboratories at Fort Belvoir, Virginia, less than 20 miles from Washington. The construction and operation to date has been funded jointly by the Atomic Energy Commission (AEC) and Department of Defense. Electricity was first generated on April 15, 1957. After the successful completion of a six-month operating test on January 24, 1958, Alco Products Incorporated, Schenectady, New York, the construction contractor, commenced operating the plant under an AEC contract which is now administered by New York Operations Office. In this capacity Alco Products, Incorporated, is responsible for the management of radioactive wastes at the APPR-1.

Although no intricate radioactive waste disposal system as such is provided at the APPR-1, provisions are made for collecting, handling and disposing of small quantities of waste during routine operations and for detecting, collecting and storing large volumes of liquid waste under abnormal conditions. This report discusses environmental monitoring and the management of gaseous, liquid and solid wastes as they apply to the APPR-1.

Environmental Radiological Monitoring

Prior to the beginning of construction at the APPR-1 an environmental monitoring program was undertaken to provide site information on activity levels before and during operation. This program is conducted by the Sanitary Engineering Branch, of Engineer Research and Development Laboratories. Continuous monitoring of dust collected on gummed paper and periodic sampling of river bottom sediment, cooling water influent and liquid waste water discharge are the major areas of emphasis in this program.

Effluent monitors are provided in the seal pit and in the stack to continuously monitor and sample all water and air as they leave the APPR-1 site. Any abnormal changes in the effluents are detected and the operating crew is alerted so that corrective action can be taken immediately. The samples are evaluated quantitatively in the laboratory to determine average alpha and beta-gamma concentrations during the period of collection. Average concentrations are maintained at 10% of the values stated in the National Bureau of Standards Handbook No. 52, when the isotopic content of the effluent is known. When the specific isotopic content of the effluent is not known, the maximum permissible average activity concentration in effluents is as follows:

Activity	Air (μc/cc)	Water (μc/cc)
Alpha	5×10^{-12}	1×10^{-7}
Beta-Gamma	1×10^{-9}	1×10^{-7}

The above values are established by AEC Regulation "Standards for Protection Against Radiation - Appendix B" dated January 1, 1958. The average concentration values are to be in addition to the normal existing background levels. Thus, the effluent monitors provide a means of insuring that the gaseous and liquid radioactive wastes are being managed properly in the APPR-1.

Gaseous Waste Products

The activity of gaseous waste products generated at the APPR-1 in 20 months of operation generally has been within tolerances for discharge through the stack to the atmosphere. The stack effluent air monitor provides a continuous check of this discharge of gaseous waste. The vent can be closed to prevent the escape of gases should the activity be above tolerances. No other management of radioactive gaseous wastes has been necessary.

Liquid Waste Products

At the APPR-1 process water which has a level of activity of less than $1x10^{-7}$ μc/cc is discharged to the seal pit and then discharges by gravity to Gunston Cove and the Potomac River. Water having an activity level greater than $1x10^{-7}$ μc/cc is treated as contaminated waste. A 5000 gallon hot waste tank and two 280 gallon laboratory waste tanks are provided for the temporary storage of these liquid wastes.

Hot Waste Tank

The hot waste tank receives radioactive water from several sources. Blowdown cooling water from the blowdown cooler and space cooler is normally discharged to the seal pit; however, should its activity be above tolerances a radiation monitor automatically operates a three-way control valve to divert the water to the hot waste tank. Primary coolant blowdown and seal leakage are automatically diverted to the hot waste tank by a radiation monitor working in conjunction with a three-way control valve if the activity of the water is too high to permit its passage through the unshielded primary coolant make-up tank, pumps and lines. If the secondary system blowdown becomes radioactive, it can be diverted to the hot waste tank, also. Spilled water in the demineralizer trench drain can be drained to the hot waste tank if it is contaminated. A schematic flow diagram of those waste disposal lines is found on page 5. In addition, cooling water from the primary coolant pumps and the pressure vessel and water from the vapor container sump can be directed to the hot waste tank.

The 5000 gallon hot waste tank, situated underground and shielded with concrete, is cylindrical with ellipsoidal heads. It is constructed of carbon steel, is 7 feet in outside diameter and 18 feet long and has a wall thickness of 3/8 inch. The tank is fitted with underground pipe connections which provide for the delivery of contaminated waste as mentioned above, venting gases to the stack,

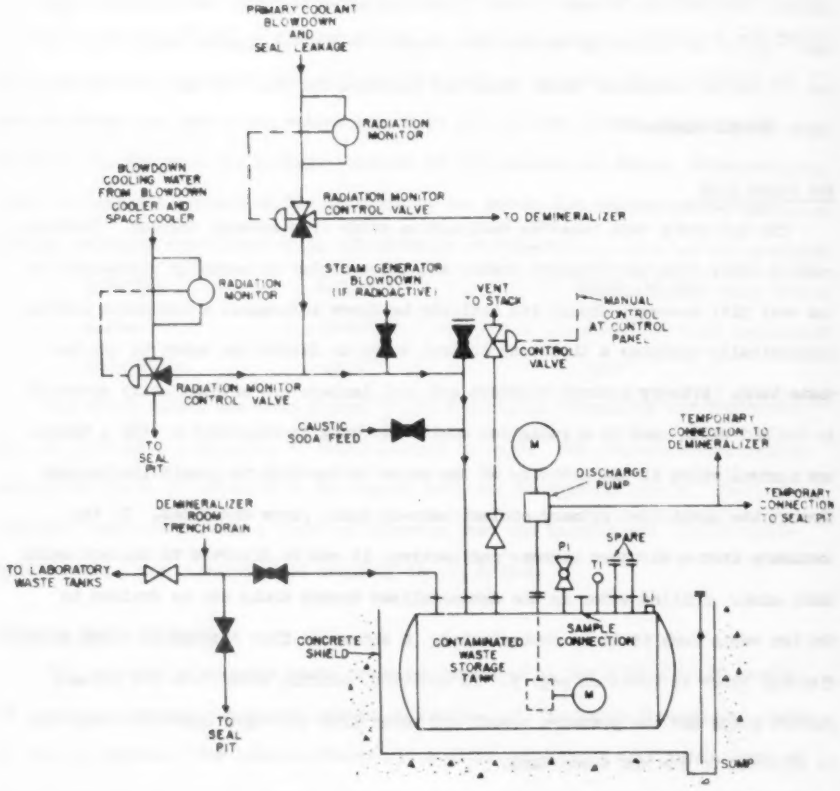

Figure 1 Contaminated Waste Disposal System, Schematic Flow Diagram (Ref 1:7)

discharging the tank, feeding caustic soda to the tank to inhibit corrosion, and pressure, temperature and level indicator attachments. A pipe from the tank to the ground surface permits the removal of samples for analysis. Two centrifugal-type tank discharge pumps, driven by 1/8 hp, single phase induction motors, are connected in series. One of the pumps is installed inside the tank and discharges to the suction of the second pump which is located outside of the tank.

Contaminated waste is held in the tank until short lived nuclides have decayed sufficiently to permit safe discharge. By means of temporary connections, the waste may be circulated through the demineralizers one or more times prior to discharge to the seal pit. In 1957 a total of 8.1 millicuries of liquid radioactive waste was discharged at the APPR-1. The quantities of radioactive waste discharged from the hot waste tank in 1958 may be found in the table on page 7.

Laboratory Waste Tank

The two 280 gallon underground laboratory waste tanks, "A " and "B", are each provided with remote level indicators calibrated to read from 0 to 47 inches. Drain lines from the hot sink in the laboratory and the sample drain in the demineralizer room run to these waste tanks. In addition, the washing machine discharge may be directed to the laboratory waste tanks in lieu of the plant septic tank if necessary. Health physics personnel sample the water in the waste tanks to insure safe discharge to the seal pit by means of a water ejector. One tank is used at a time leaving the other tank for standby or holding contaminated waste until its activity has decayed sufficiently to permit discharge.

For purposes of discussion the use of the laboratory waste tanks for the collection, handling and disposal of radioactive liquid wastes will be considered in two categories, high and low level. The former includes samples which contribute significantly to the total activity in the laboratory waste tanks but

Radioactive Liquid Waste Discharge From The APPR-1 in 1958

Date (1958)	Tank	Quantity (gal)	Activity Level uc/cc	Microcuries Discharged
17 January	Lab A	230	7.4×10^{-4}	643
"	Lab B	250	7.45×10^{-4}	705
23 February	Lab B	260	5.4×10^{-4}	530
14 March	Hot Waste	500	6.8×10^{-4}	1285
27 March	Lab A	200	5×10^{-3}	3780
"	Lab B	250	2×10^{-3}	1900
3 April	Hot Waste	800	9.65×10^{-4}	2920
2 May	Lab A	250	2.9×10^{-3}	2746
"	Lab B	250	2.3×10^{-3}	2173
3 May	Hot Waste	1500	4.5×10^{-4}	2552
16 June	Lab B	240	1.49×10^{-3}	1305
23 July	Lab A	260	8.5×10^{-4}	835
"	Hot Waste	1200	1.88×10^{-5}	85
6 August	Lab A	240	2.62×10^{-4}	235
"	Lab B	280	8.65×10^{-4}	915
26 August	Hot Waste	600	7.15×10^{-4}	1615
11 September	Lab B	240	1.05×10^{-3}	954
27 September	Lab B	280	1.15×10^{-3}	1220
3 November	Lab A	280	1.56×10^{-3}	1650
4 November	Lab B	225	4.48×10^{-4}	378
15 December	Lab B	175	1.99×10^{-3}	1320
24 December	Lab A	260	1.64×10^{-4}	161
27 December	Hot Waste	1200	1.4×10^{-3}	6360
28 December	Hot Waste	1200	1.4×10^{-3}	6360
		11,170		42,627

are relatively low in activity level compared to other waste management operations. The low level wastes are generally less than $1x10^{-7}$ uc/cc.

High Level Radioactive Liquid Waste

One of the principle sources of liquid waste is from water taken from the primary side of the reactor coolant system. These samples are taken daily as part of the routine plant chemistry analysis. In orders of magnitude, the concentration of this radioactive water is 10^{-1} uc/cc on the upstream side of the demineralizer and 10^{-3} uc/cc on the downstream side of the demineralizer.

The specific analyses, quantities, handling and disposal are as follows:

1. DF-30 pH and Conductivity Tests of Primary Blowdown.

   The samples are collected in 250 ml Berzelius Beakers. Approximately 200 ml of primary blowdown upstream of the demineralizers and 200 ml of primary blowdown downstream of the demineralizers are collected daily. The technician is required to wear a laboratory coat, rubber gloves and a respirator while sampling. The two (2) 200 ml samples are brought into the laboratory where 2 ml of each sample is pipetted into a 3 ml planchette for determination of the DF-30. (DF-30 is the demineralizer factor taken 30 minutes after sampling and gives an indication of effectiveness of the demineralizer). Planchetted samples are taken to dryness on an electric hotplate enclosed in a radioactive material exhaust hood and counted with a scaler. Sample planchettes are then placed in solid contaminated waste cans for later disposal. Conductivity and pH measurements are made with the remainder of the two (2) 200 ml samples. After these tests the remainder of the liquid sample is discarded into a contaminated waste drain located in the exhaust hood sink which empties into the laboratory waste tanks.

2. Oxygen Samples of Primary Blowdown and Primary Make-up Tank.
One 500 ml sample is taken from the Primary Blowdown upstream of the demineralizer and one from the Primary Make-up Tank. These samples are collected daily in two (2) 500 ml McLean flasks to determine dissolved O_2 content in the primary water. Technicians are required to wear same protective clothing as in No. 1 above. Five volumes of water are passed through each McLean flask while connected to the sample point. Overflow water discharges to the laboratory waste tanks. When the samples have been collected, the McLean flasks are then removed to the laboratory where reagents are added under the radioactive material exhaust hood. The samples are then emptied from the flasks to open white porcelain dishes where a titration operation is performed. Water from the dishes is emptied into the contaminated waste drain after all analyses are completed.

3. Hydrogen Samples of Primary Blowdown and Primary Make-up Tank.
These samples are taken in the same manner as described in No. 2 above, except that only one (1) 500 ml McLean flask is used. Samples are brought to the laboratory where gas is removed in a standard Orsat apparatus. The liquid sample, after the gas is removed for analysis, is emptied into laboratory waste tank drains.

4. Samples from the Hot Waste and Laboratory Waste Tanks.
High level radioactive waste (10^{-3} uc/cc) from the laboratory waste retention tank and the hot waste tank is also analyzed in the laboratory. The collection of these samples involves connecting a small suction pump to a sampling line which is lowered into the waste tanks. A 25 ml aliquot of this material is evaporated to dryness on a planchette in the laboratory

using the same precautions as previously mentioned. These samples after being counted are disposed of in the waste container located in the laboratory. The flasks used for collecting these samples are rinsed and washed in the laboratory exhaust hood sink which drains to the laboratory waste tank. After the first rinsing and washing the glassware is further washed in the laboratory sink which drains to the plant septic tank.

The laboratory waste tanks receive all of the waste from the chemical laboratory which is low enough in radioactivity levels to permit discharge into the Potomac River under controlled conditions. To date, no water of activity high enough to prohibit controlled release into the Potomac River has been accumulated in these tanks. The table on page 7 shows the total liquid waste accumulated as a result of operations at the APPR-1 for the year 1958. It is to be noted that the total activity discharged to the river is quite small for a period in which the plant output was 6.75 MWY. The laboratory waste tanks are discharged to the river in such a manner that effluent from the seal pit does not exceed 1×10^{-7} uc/cc above background. The effluent is monitored using a G-M type instrument as well as by direct laboratory analysis of a continuous drip sample of the plant waste water before it is diluted with river water.

Low Level Radioactive Liquid Wastes

All water entering and leaving the plant is monitored on a daily routine schedule in order to insure no release of radioactive material above tolerances into the environment. Three samples are taken daily for this purpose and include analyses of incoming river water and outflowing water taken from the seal pit before discharge into the river. The seal pit is monitored on a continuous basis by a Nuclear Measurements Corporation instrument using a

G-M tube detector. The upper limit for radioactivity in the effluent is $1x10^{-7}$ uc/cc and, therefore, extreme care must be exercised to prevent contamination of test equipment. These samples are collected in glassware set aside for this purpose in order to eliminate the possibility of cross contamination. A 500 ml sample is evaporated to dryness on a planchette and counted for radioactivity on a conventional decade scaler utilizing a mica end window beta-gamma detector. The glassware used in the collection and evaporation of these samples is rinsed in the exhaust hood sink and then washed in the laboratory sink. The planchettes used in counting these samples are discarded as solid radioactive waste. Up to the present time there have been no problems from the laundering of clothing at the plant and all laundry water has been discharged to the plant septic tank. All clothing is monitored before washing and any contaminated clothing is discarded as solid radioactive waste.

The activity of the liquid effluent is generally less than $1x10^{-7}$ uc/cc though on occasion higher levels have been detected in the cooling water influent (Ref 2). Based upon the data obtained by the Sanitary Engineering Branch, it appears that these conditions are due to radiation fallout which corresponds to nuclear detonation testing. During these periods there is no discharge of liquid wastes into Gunston Cove.

Solid Radioactive Waste

Under normal operating conditions, the bulk of solid radioactive waste is accumulated as a result of the plant chemical laboratory activities. The waste consists mainly of discarded or broken contaminated glassware, planchettes and paper towels. These materials are collected in metal waste containers provided for this purpose. The containers consist of a covered metal garbage can lined with a disposable bag. The level of radiation is not permitted to exceed 20 mr/hr

at the surface of the container. Any article which exceeds this value is handled separately in a special container. The solid waste is stored in 5 gallon steel cans in a waste retention area. These cans are used for shipping this waste to an outside contractor who undertakes final disposal. At the time of shipping the outside of the container is monitored to insure conformance with Interstate Commerce Commission shipping regulations.

The greatest source of radioactivity is found in the demineralizers used for the removal of corrosion products from the primary system coolant. The activity in this system occurs as a result of induced radiation in the circulating corrosion products when they pass thru areas having high neutron fluxes. Presently, the only data available on activity levels is from a demineralizer that was in operation for the period of one year. During this time the plant was operating at design power about 62% of the time. At the time of removal the dose on the surface of the demineralizer was about 90 R/hr. It was inspected and disposed of by Oak Ridge National Laboratory (ORNL). The demineralizers are contained in lead shipping casks during operation. Upon depletion of the resin the cover of the cask is secured and the cask is disposed of at ORNL. The minimum life of the demineralizer is approximately one year.

It is expected that more serious problems in solid waste disposal will be encountered at such time as major changes in the primary system are made. In the near future a control rod in the first core in the APPR-1 will be removed for physical and metallurgical examination at ORNL. In September or October it is expected that the entire core and the steam generator will be replaced. Pending results of Research and Development tasks on the buildup of radioactive nuclides in the primary system and procedures for decontaminating the primary system (Steam generator), the extent of the problem of disposing of the steam generator will be more clearly established.

Economics of Waste Management Operations

Because of the small magnitude of waste management operations at the APPR-1, costs related to this phase of plant operation are not accounted for separately. It is possible, however, to estimate the additional costs and/or labor introduced by the presence of radioactive wastes. The level of effort in the environmental monitoring program conducted by the Sanitary Engineering Branch, ERDL is approximately $10,000 per year. At the APPR-1, one-half man year is more than adequate to perform the health physics operations associated with waste management. The work associated with the disposal of wastes is performed by the operating crew and is incidental to other operations. By use of automatic instrumentation it may be possible to reduce the health physics effort in APPR type reactors. This will be investigated under a research and development task in calendar year 1960 for improvements in instrumentation for the APPR-1.

Summary

The stack effluent air monitor has been the only control necessary for the discharging of gaseous waste products at the APPR-1. A 5000 gallon hot waste tank and two 280 gallon laboratory waste tanks provide a means for storing radioactive liquids until the short lived activity decays sufficiently to permit discharge to the Potomac River. Water samples taken for routine plant chemistry operations are the greatest source of radioactive liquid wastes which in turn are the greatest problem in waste management at the APPR-1. In 20 months of operation it has been possible always to discharge the gaseous and liquid wastes to the environment. Occassionally it has been necessary to delay the discharge of liquid wastes because of a high specific activity in the river water. The total activity of the liquid wastes discharged to date is approximately 50.8 millicuries. Solid radioactive wastes have been handled in a routine manner. The replacement of a

demineralizer has been the only shipment of waste of a high specific activity.

The methods of managing radioactive wastes at the APPR-1 are relatively simple and inexpensive.

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RADIOACTIVE WASTES AT THE CONVAIR-FORT WORTH PLANT

By B. P. Leonard[1] and F. L. Paschal[2]

INTRODUCTION

Convair-Fort Worth, a Division of General Dynamics, has been operating two nuclear reactors since 1953. These reactors are being used in a research and development program to investigate and solve the shielding and radiation effects problems which arise in an aircraft nuclear program. The plant is also engaged in the primary activity of fabricating new, high-performance airborne weapons systems and in modifying and modernizing existing aircraft. It is important to note that, in bulk, the aircraft programs have proven to be greater sources of radioactive wastes than have the nuclear research activities. This quantity and types of radioactive wastes generated as a result of all of these programs have posed new and different problems in waste disposal, some of which have not yet been satisfactorily resolved.

Waste disposal at Convair is carried out under four sets of regulations:

1. Atomic Energy Commission Regulations.
2. Regulations of the Health Department of the State of Texas.
3. Applicable Air Force technical orders.
4. Applicable handbooks as issued by the National Bureau of Standards, U.S. Department of Commerce.

As a general policy, line departments in the plant perform their regular functions whether the material they handle is radioactive or not. The Health Physics Group has the responsibility of maintaining surveillance over the line departments. The salvage and disposal section segregates and processes radioactive wastes in a separate disposal facility.

SOURCES OF RADIOACTIVE WASTES

Radioactive wastes at the Fort Worth plant are generated primarily from three activities:

1. Nuclear reactor operations.
2. Aircraft maintenance and modification.
3. Airborne weapons systems fabrication.

In each activity the wastes are generated to some extent in gaseous, liquid, and solid states.

When neutrons produced by operating reactors are absorbed, they cause the absorbing material to become radioactive. The reactor operations at Convair-Fort Worth have led to the activation of many different types of materials, such as: air and other gases; water, jet fuel, lubricants, and other liquids; and aircraft materials, components, and systems.[3]

Since nuclear weapons testing began, aircraft, particularly those which fly at high altitudes, have become contaminated to some extent by radioactive bomb debris in the atmosphere. In some cases, the contamination is so great that decontamination or close monitoring of maintenance and modification opera-

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Dr. Leonard assumed duties at Convair with the aircraft nuclear propulsion program in 1953. His initial efforts were directed toward planning and supervising an experimental shielding program. Since 1955, when he was appointed Chief of the nuclear research and development organization at Convair, Dr. Leonard has directed research in radiation shielding and radiation effects on materials and equipment. Associated with this effort has been the design and fabrication of research reactors, nuclear instrumentation, experimental equipment, remote-handling equipment, and the study of radiological hazards. He has presented several papers on the calculation of the spread of released radioactivity in the air.

[2] Frank Paschal organized the health physics program at Convair-Fort Worth in 1953 and is serving at present as the health physics program administrator.

He attended Washington and Lee University and, during the war, served in the U.S. Navy. He was graduated from the University of Texas in 1947 and did graduate work at Trinity University.

In 1948 he was granted an AEC fellowship in radiological physics. After receiving health physics training in the first fellowship class at ORNL, he did health physics work from 1950 to 1953 at Westinghouse, Atomic Power Division, Pittsburgh.

[3] When the nuclear test airplane was carrying an operating reactor, the entire airplane was activated to some extent, although not to levels which would create significant handling problems.

tions is required. It is usually faster and cheaper to decontaminate the airplane. Wastes generated during the decontamination of airplanes include:

Airborne activity in steam and water vapor.
Contaminated water and cleaning solutions.
Contaminated clothing, rags, brushes, and other cleaning equipment.

The high performance of new weapons systems being fabricated by Convair has desulted in the use of special materials. One of these is a magnesium-thorium alloy, which contains the naturally radioactive element thorium. Radioactive wastes which are generated during fabrication operations using this alloy include:

1. Airborne radioactivity, both particulate and gaseous.
2. Contaminated liquids used in fabricating and chemical processing.
3. Solid materials in the form of shavings, chips, cutoffs, etc.

CONTROL AND HANDLING

The control of radioactivity at Convair is complete in that radiation levels and the amount and location of radioactive materials are known at all times. Control begins the moment radioactive materials are brought into or produced at the plant. Control ends only after the radioactive materials are no longer being used and—

1. The levels of radiation are below the maximum limits established for nonoccupational exposures;
2. Proper precautions have been taken to prevent the accumulation of radioactive materials in a way that might subject nonoccupational persons to radiation in excess of established limits.

Handling and monitoring procedures for radioactive materials in the plant differ, depending upon the type and the level of radiation. These procedures may be divided into two general categories: those for radiation levels above established limits for nonoccupational personnel; and those for radiation levels below those limits. Access to areas in which high radiation levels exist is restricted to those persons who must work in the areas and each individual in these areas is personally monitored. Freer access is permitted into those areas where low radiation levels exist, but the areas themselves are carefully monitored to insure that no buildup of radioactive materials occurs. In order to verify the adequacy of the control measures, routine investigations are made to determine the levels of radioactivity in air, soil, water, vegetation, and other materials. Once each month water samples are collected at 28 stations, soil samples at 31, and vegetation samples at 19. The radioactivity levels of these samples are compared to those established prior to nuclear operations. In addition, the air, the water at the Lake Worth input to the city water system, and the effluent from Convair's sanitary sewer are monitored continuously. These monitoring services—of areas, personnel, and environment—are provided by a health physics staff of 28 people.

Figure 1 is a picture of the continuous water monitor located at the Lake Worth input to the city of Fort Worth waterworks.

COLLECTION

The method of collecting and disposing of radioactive wastes depends upon the physical and chemical nature of the wastes and upon the concentration of radioactive materials in them.

Nuclear-reactor operations result in the accumulation of liquid and solid radioactive wastes at the operations and laboratory areas. Liquid wastes generated in the reactor operations area are collected in three holding tanks having a total capacity of 19,000 gallons. After irradiation, aircraft parts, materials, and components are taken to the Irradiated Materials Laboratory for radiation effects testing. Liquid wastes generated in the Irradiated Materials Laboratory and in the decontamination laundry (fig. 2) are collected in the three holding tanks shown in figure 3.

Solid wastes from both the operations and the laboratory areas are collected in specially marked barrels in combustible and noncombustible categories. These barrels are stored in a waste-disposal building located at the decontamination facility.

The decontamination facility consists of a paved area of approximately 10,000 square feet. Outlets are available for compressed air, water, steam, and electricity. The facility has four retaining tanks with a total storage capacity of 80,000 gallons and a drainage overflow pit with a capacity of approximately

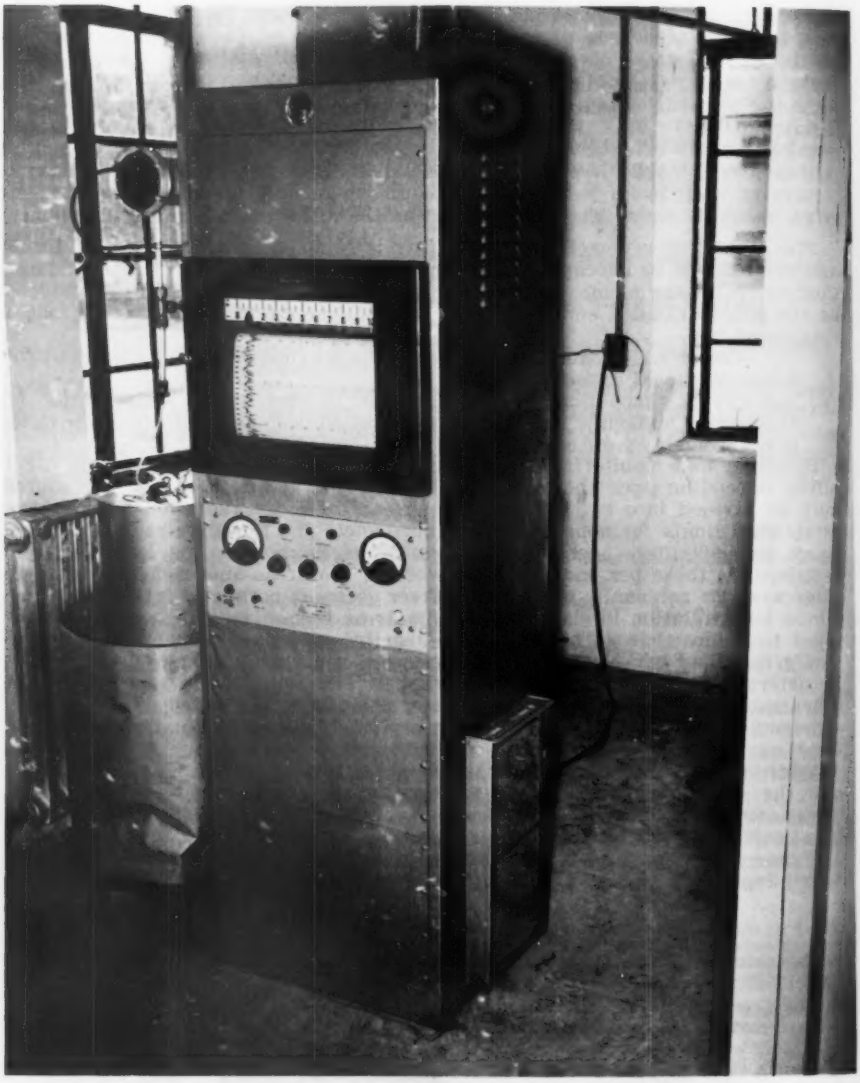


FIGURE 1.—Water radioactivity monitor at Lake Worth.

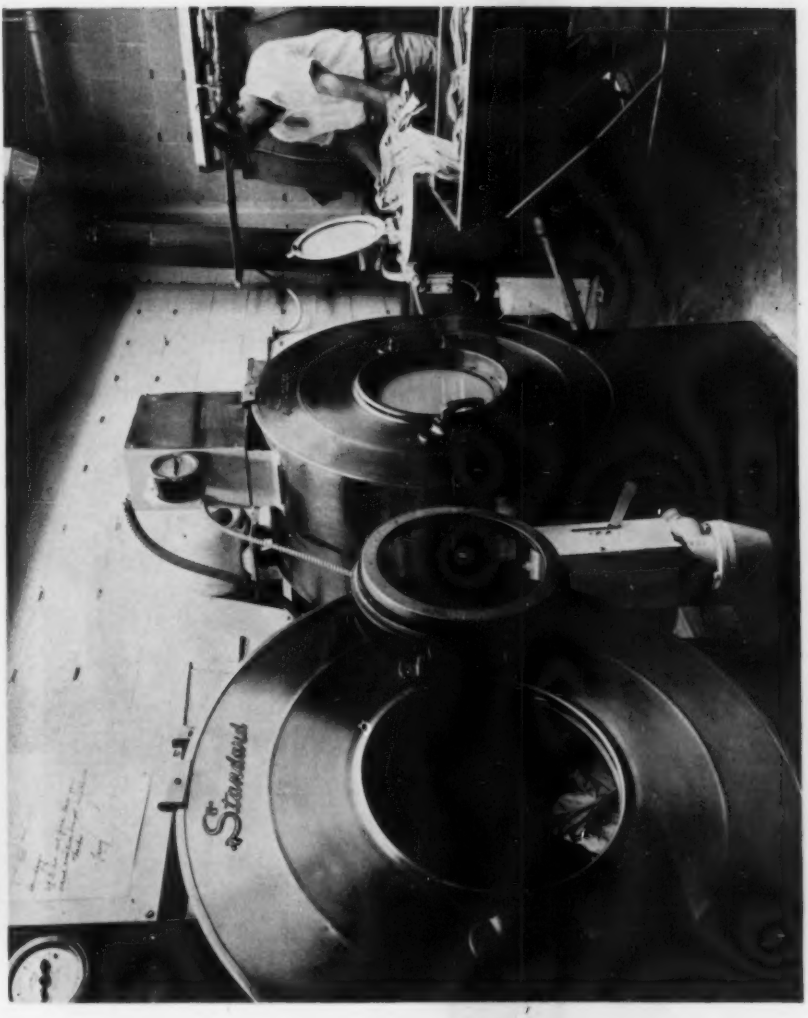


FIGURE 2.—The decontamination laundry in the Irradiated Materials Laboratory.

FIGURE 3.—Collection tanks at the irradiated materials laboratory.

FIGURE 4.—Waste disposal and decontamination facilities.

FIGURE 3.—Collection tanks at the irradiated materials laboratory.

FIGURE 5.—Sewage radioactivity monitor.

FIGURE 5.—Sewage radioactivity monitor.

FIGURE 6.—Incinerator for radioactive combustibles.

450,000 gallons. It also has change-room facilities, a storage building for materials and radioactive waste, and an incinerator for burning low-level radioactive waste matter. A view of the facility is shown in Figure 4.

Solid wastes generated in magnesium-thorium fabrication operations are collected in specially marked barrels and moved to the waste-disposal building at the decontamination facility. Noncorrosive contaminated liquid wastes are also collected in specially marked metal drums and moved to the waste disposal building. Corrosive contaminated liquid wastes are collected in 5-gallon glass bottles placed inside metal containers for movement to the waste-disposal building.

DISPOSAL

The basic methods for disposing of radioactive wastes include release of liquids into the sewer system, burning, reclamation, and storage and burial.

Chemically nontoxic, water-soluble liquids are released into the sewer system only after the wastes are sufficiently diluted to reduce the concentration of radioactivity below established levels for drinking water. To assure that dilution is adequate, a floating monitor (fig. 5) is located at the sewer exit from the nuclear research area.

Burning of combustible materials is done only with materials having low levels of activity so that the resulting air contamination is below established levels for breathing by nonoccupational personnel. Burning operations are continuously monitored. Figure 6 shows the incinerator used exclusively for the burning of radioactive combustible wastes. The stairs are rolled up to the stack and used in monitoring the effluent.

Magnesium-thorium wastes generated in fabrication operations are sold as scrap to authorized companies for reprocessing.

Wastes not falling into the above categories are sealed in metal drums and stored until they decay to a point where they may be disposed of in one of the manners described above. In cases where this is impractical, the wastes are sealed in metal drums and shipped to authorized receivers for final disposal.

The following table summarizes the amount of radioactive wastes disposed of by Convair, Fort Worth, between January 1954 and October 1958:

TABLE I.—*Disposition of radioactive wastes at Convair, Fort Worth (January 1954 to October 1958)*

Source	To sewers microcuries	Solids burned (cubic feet)	Liquids burned (gallons)	Shipped or stored (cubic feet)	Resold (pounds)
Nuclear program	93	1,340	825	1,768	----------
Decontamination	1.1	350	2,600	2,170	----------
Fabrication	----------	----------	----------	1,466	6,050

On hand at present are approximately 1,200 cubic feet of waste. The fact that much of this has been generated by the fabrication programs makes it evident that, even though not engaged in nuclear activities, many aircraft plants have or will have problems in handling and disposing of radioactive wastes.

SPECIAL PROBLEMS

The magnesium-thorium alloy mentioned previously is a source of the radioactive gas Rn^{220}, one of the daughter products of the decaying thorium. It in turn has several radioactive daughter products. Because of this, there is a problem in storing the alloy in large quantities. Also, since certain machining operations produce small particles which can become airborne, good ventilation of the fabrication areas is required. Not only does instrumentation for detecting and measuring very low levels of thorium in air leave much to be desired, there is at present insufficient data on biological effects on which to base a sound selection of the maximum permissible concentration in air for thorium and its daughter products. Suggestions for the maximum permissible level in the air range all the way from 1 to 270 micrograms per cubic meter (table II).

TABLE II.—*Maximum permissible levels of thorium in air*

Authority	Maximum level ($\mu g/m^3$)
Federal Register	90
ICRP	270
Dow Bulletin No. 141-179	100
HW-40105	3
KAPL-896	1

As larger quantities of the magnesium-thorium alloy are used and as fabrication operations become more extensive, the radioactivity problems connected with it will become more acute.

The contamination of aircraft by radioactive debris in the air from nuclear weapons tests is becoming a widespread problem; to some extent every flying aircraft is subject to this contamination. Problems in this area will have their effect, not only on aircraft manufacturers, but also on military, commercial, and private aviation. Since the activity is loose, it can be rubbed off onto clothing and skin of personnel working on the airplane. Systems or components of contaminated aircraft may be returned to vendors and subcontractors for rework, thus finding their way into many different shops located throughout the country. Some aircraft now undergoing modification are contaminated to such an extent that they and their components must be handled as radioactive materials. Because of the size of the equipment involved, decontamination is generally the simplest and cheapest handling method.

This problem is complicated by the fact that to date there exist no national standards of permissible contamination levels. Further, the methods for determining the contamination of an aircraft are somewhat uncertain. The general method is to wipe or "swipe" a prescribed area with a piece of soft paper and then to determine the amount of activity removed. Different people performing these tests in the same areas may obtain considerably different results. In addition, there is generally no detailed knowledge of the precise nature of the radioactive material present. This is rather meager information on which to base a decision to decontaminate an aircraft and to institute special procedures for handling and disposal.

Many problems are encountered in the handling, control, and disposal of radioactive wastes. The two mentioned above are singled out for attention because they are basic problems which are becoming rather widespread and which require considerably more investigation.

CONCLUSIONS

Most of the criteria and methods for the safe handling, control, and disposal of radioactive wastes at Convair, Fort Worth, are well established. An unblemished safety record attests to this. However, additional basic information and an improvement in techniques are needed in some cases.

Many activities not normally considered to be nuclear in nature will be confronted with radioactive waste problems in the future. It is important that they be anticipated in advance so that, when they occur, appropriate measures of control may be exercised.

The increased production and use of radioactive materials will make the waste disposal problem increasingly difficult. Organizations concerned with radioactive wastes must become increasingly aware of the disposal activities of other organizations, so that the cumulative disposal of radioactive wastes is carried out in such a way as not to be injurious to the well-being of the general public.

RADIOACTIVE WASTE DISPOSAL
AT THE BETTIS PLANT

Prepared For

Pittsburgh Naval Reactors Operations Office
of the
United States Atomic Energy Commission
by the
Bettis Plant Industrial Hygiene Department

WASTE DISPOSAL

I. GENERAL

All of the liquid, solid and airborne radioactive wastes, which are generated at the Bettis Plant, are collected, monitored, processed and disposed of in accordance with prescribed limits and procedures.

Special drain lines carry the radioactively contaminated liquid wastes (water) to 1500-gallon retention tanks for sampling and analysis. If the concentration of radioactivity in the liquid is below specified limits, the water is dumped to the storm sewer and is discharged with the other water effluent from the plant. Waste which exceeds specified limits of radioactivity is processed. Processing consists of concentration, filtration or ion-exchange.

Concentration is accomplished by boiling the raw liquid, thus removing the bulk of the water and leaving the radioactive material in the concentrate. The concentrate is then mixed with an absorbent material and packed into 30-gallon drums. These drums are sealed in concrete inside 55-gallon drums and sent to the ocean for burial.

Filtration involves sending the liquid through mechnaical filters to remove the radioactive particles entrained in the water. In this way the concentration of radioactivity is reduced to a point where the liquid can be released to the storm sewer.

Ion-exchange is used on a relatively small volume of water which contains a low concentration of dissolved solids.

The total liquid effluent from the plant is continuously sampled and analyses are made of these samples to insure that the concentration of radioactivity in the effluent is controlled within permissible limits.

A wide variety of solid radioactive wastes are generated throughout the Bettis Plant. The radiation intensity varies between a fraction of a milliroentgen per hour on surface-contaminated solids to several hundred roentgens per hour from discarded irradiated specimens. All solid radioactive wastes can be classified as combustibles or non-combustibles. Radioactive combustibles are reduced to ash in a specially designed incinerator. Ashes from this process are packaged and sent to the sea for burial. The smoke is passed through a wet collector (Rotoclone) where the bulk of the particulate material is collected in water. The cleaned exhaust is then released to the atmosphere.

Radioactive non-combustible wastes are sealed in concrete inside 55-gallon drums and sent to the ocean for sea burial. If the waste is highly radioactive, it is first placed inside lead shields which are then sealed in concrete inside 55-gallon drums. All drums are packaged to meet ICC shipping regulations.

The discharge of air which might contain radioactive dusts, fumes and mists from plant processes is carefully controlled. Wet collectors, electrostatic precipitators, high efficiency filters and various dry filter media are used

to remove the radioactive particulates from the air before it is discharged to the atmosphere. These discharges of air are sampled on a routine basis to assure that the average concentrations of radioactive contaminants are maintained within acceptable limits.

A report describing radioactive waste disposal practices at the Bettis Plant was prepared early in 1958 for the U. S. Atomic Energy Commission, Pittsburgh Naval Reactors Operations Office (PNROO), who, in turn, submitted it to the Commonwealth of Pennsylvania Department of Health. Quarterly reports presenting data on liquid waste discharges from the plant are also submitted to them through PNROO.

II. LIQUID WASTES

A. Sources and Type of Activity

The radioactivity in the liquids (mainly water) at Bettis Plant comes from a variety of laboratory and plant operations. Typical sources of this contaminated water are: shops and laboratories where water is used for the decontamination of facilities and equipment, the decontamination of components which have been removed from operating reactors, process water from the fabrication and analyses of fuel materials, water used for laundering radioactively contaminated protective clothing, water from wet dust collectors on processes and operations which generate airborne radioactive contamination, and moderating water used in critical experiment facilities.

Through surveilance of the laboratory and shop operations, the locations which handle radioactive materials are identified and work which might produce contamination is restricted to those locations where facilities for proper control are provided. Special care is taken to see that processes, which might release material having a low Permissible Concentration, are performed in well protected locations. A list of the principle existing isotopes which are handled in the Bettis Plant Operations is presented in Table I.

TABLE 1

LIST OF RADIOACTIVE MATERIALS EXISTING AT BETTIS PLANT

1. Fuel Materials: Natural and Enriched Uranium and Uranium Oxide

2. Products From Irradiation of Fuel Materials (mainly mixed fission products age 3 mos. to 3 yrs.):

Sr-89	Nb-95	Cs-137	Ce-144
Sr-90	Ru-103	Ba-140	Pr-144
Y-90	Rh-103	La-140	Pm-147
Y-91	Ru-106	Ce-141	Pu-239
Zr-95	Rh-106	Pr-143	Pu-240
			Pu 241

3. Laboratory Isotopes and Radioactive Corrosion Products:

Na-22	Co-58	Ag-110	Cs-134
K-40	Fe-59	In-114	Cs-136
Cr-51	Co-60	Cd-115	Hf-175
Mn-54	Mo-99	I-131	Hf-181

B. Collection, Monitoring and Processing

1. Drain Systems

There are five (5) types of drain systems within the plant. They are Sanitary drains, Storm drains, Special Monitor drains, Monitor drains and Accountability drains.

a. Sanitary drains carry only sanitary wastes to the Sanitary Waste Treatment Plant or to septic tanks. The effluent from the Sanitary Waste Treatment Plant empties into the Storm drain prior to its leaving the plant property.

b. Storm drains carry surface drainage, and process and laboratory waste water, which is not contaminated with radioactive materials, to the plant effluent. This drain system by-passes the Sanitary Waste Treatment Plant.

c. Special Monitor drain systems carry water, which is known to be contaminated with radioactive materials, to the Monitor Tank Rooms where it is collected in storage or retention tanks. Labeled laboratory and wash sinks, floor drains, etc., in areas where radioactive materials are used or processed, are connected to these systems.

d. Monitor drain systems carry water which is normally not contaminated with radioactive materials, but which might accidentally become contaminated. The water which enters this system from labeled sinks, floor drains, etc., is collected in storage tanks in the Monitor Tank Rooms.

e. The accountability drain system carries water which is contaminated with enriched uranium. This system is connected to all the sinks, floor drains, etc., in only one shop, where production quantities of enriched uranium alloys are fabricated. The water from this drain system is collected in separate storage tanks in the Monitor Tank Room to permit the accounting for and the possible reclamation of the enriched uranium which has a high intrinsic value.

The water collected in the Monitor and Special Monitor storage tanks is sampled, analyzed for radioactivity, and either released to the Storm drain or processed in the waste treatment facilities. The water collected in the Accountability storage tanks is sampled, analyzed for uranium alpha activity, processed in separate waste treatment facilities and the concentrated liquid is retained for possible reclamation of the enriched uranium.

The total storage capacity for Monitor and Special Monitor water, servicing four separate areas within the plant, is 76,500 gallons. This will be increased to 84,000 gallons in the near future. The total storage capacity for Accountability water is 4500 gallons.

Contaminated water which, because of its radioactive content, cannot be released to the Storm drain is treated by one of the following processes.

1. Boiler and Evaporator Liquid Waste Treatment Process

A schematic diagram of this process is shown in Figure 1. Two areas within the plant use this process, one area being equipped with three boilers and three evaporators and one area being equipped with one boiler and one evaporator. The contaminated water is concentrated and reduced in volume by evaporation in the vertical tube gas-fired boilers, each having an evaporating capacity of 150 to 200 gallons per hour. Entrained water or particulates are removed from the steam by a steam separator and returned to the boiler. The steam and flue gases are discharged to the atmosphere through a common stack. The stack effluent is sampled and analyzed for radioactivity.

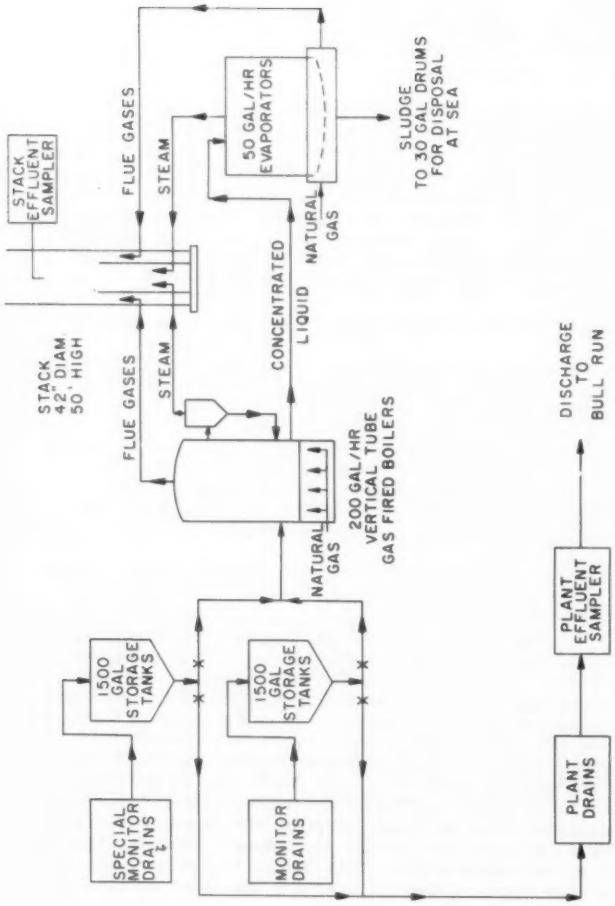
To prevent excessive deposition of solids on the tube walls of the boilers, the concentrated liquid is transferred to pot-type gas-fired evaporators for final concentration. The steam and flue gases from the evaporators are discharged to the atmosphere through the boiler stack. The concentrated liquid from the evaporators is mixed with an absorbent material and packed in 30-gallon drums for disposal at sea. This process reduces the volume of contaminated liquids by a factor of approximately 200. No liquids from this process are released to the plant liquid effluent.

2. Vapor Compression and Spray Dryer Liquid Waste Treatment Process

Three vapor compression distillation units and two spray dryers are used in one area within the plant. A schematic diagram of the process is shown in Figure 2.

The vapor compression distillation process is a method of recycling the latent heat of vaporization in the steam from the boiling solution back to the boiling solution. Such a unit, shown schematically in Figure 3, consists primarily of an auxiliary steam boiler, evaporator, steam filter, vapor compressor and heat exchanger.

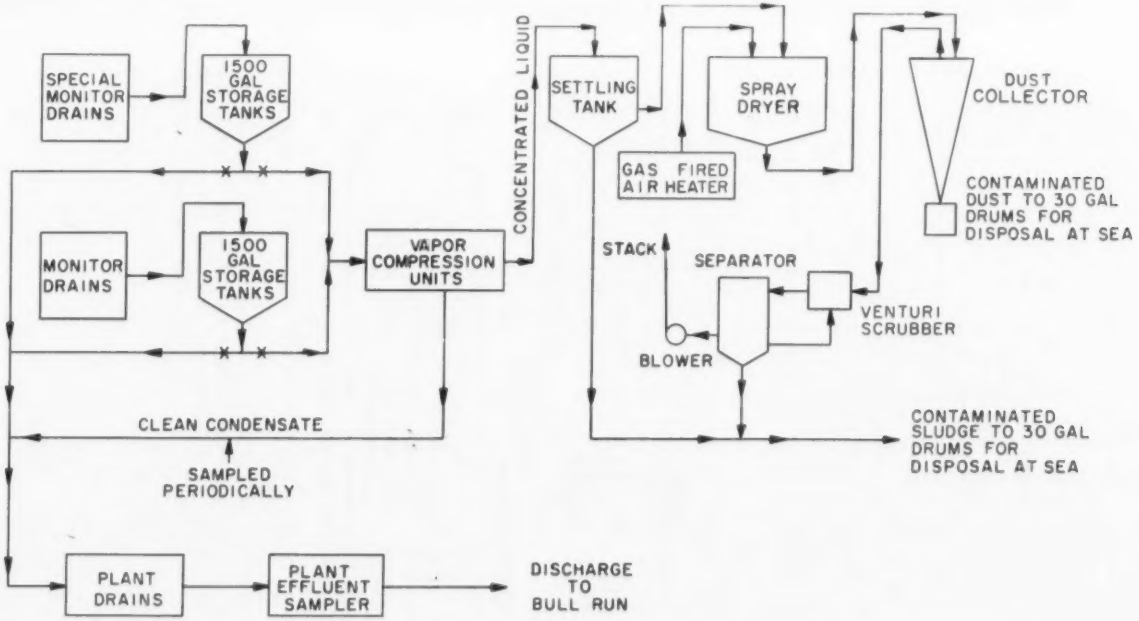
Contaminated water in the evaporator is initially brought to a boil with heat produced by an auxiliary steam boiler. The vapor compressor draws the steam from the boiling solution through the steam filter, which removes entrained solid and liquid particles, compresses it and forces it into the steam chest of the evaporator where the heat is transferred to the boiling solution. The compression of the steam raises its temperature and adds to the latent heat of the steam the amount of heat necessary to continue vaporization of the feed solution in the evaporator. Heat losses in the system are made up by the auxiliary steam boiler. The vapor compressor also reduces the pressure in the vapor head of the evaporator thereby lowering the boiling point of the feed solution. The distillate (condensate) from the steam chest of the evaporator passes through the heat exchanger to preheat the incoming feed solution and is discharged to the Storm drain. This distillate is analyzed for radioactivity as a check on the satisfactory operation of the unit.



BOILER AND EVAPORATOR
LIQUID WASTE DISPOSAL SYSTEM

FIGURE 1.

FIGURE 1.



VAPOR COMPRESSION-SPRAY DRYER
LIQUID WASTE DISPOSAL SYSTEM

FIGURE 2.

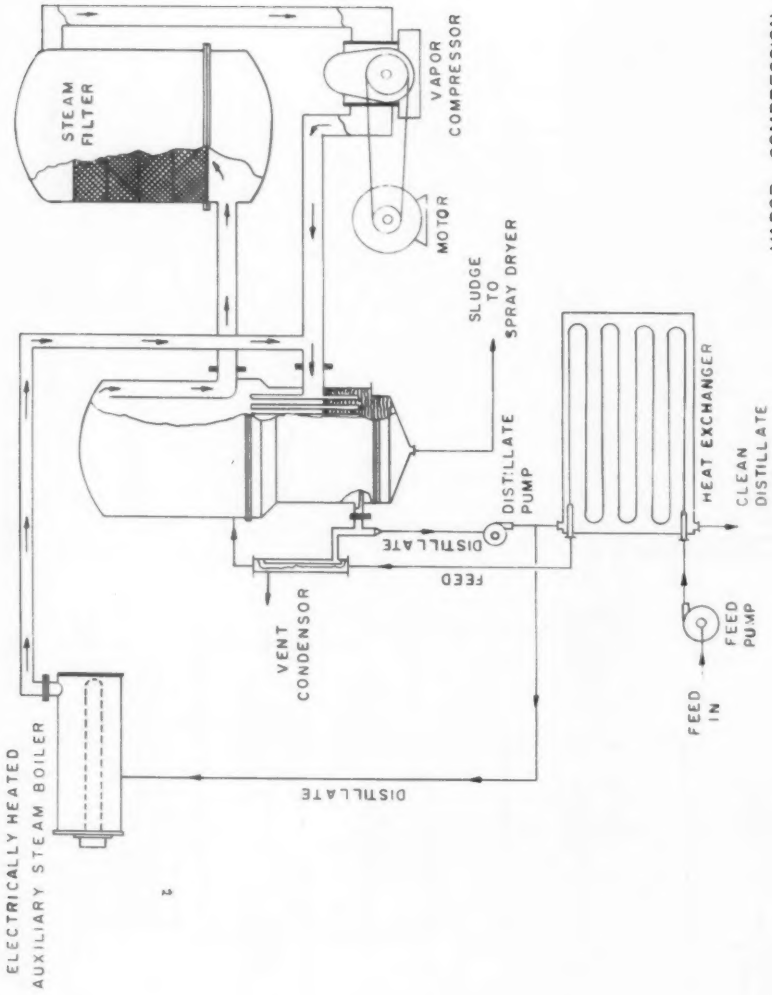

STEAM FILTER
VAPOR COMPRESSOR
MOTOR
SLUDGE TO SPRAY DRYER
DISTILLATE PUMP
HEAT EXCHANGER
CLEAN DISTILLATE OUT
DISTILLATE
FEED
VENT CONDENSOR
FEED PUMP
FEED IN
DISTILLATE
ELECTRICALLY HEATED AUXILIARY STEAM BOILER
VAPOR COMPRESSION SYSTEM

FIGURE 3.

A decontamination factor of 10^3 is realized. The concentrated liquid from the evaporator section of the vapor compression distillation unit is pumped to a settling tank where the solids are permitted to settle out. The supernatent liquid comprises the feed solution for the spray dryer and the settled solids are packed for disposal at sea.

The spray drying process is shown schematically in Figure 4. The feed solution is sprayed as a mist into the chamber of the spray dryer where the droplets are entrained in the hot air steam. At a constant air velocity, the solution feed rate and the air temperature are adjusted so that the water is evaporated from the droplets leaving only dry dust particles or agglomerates, containing the radioactive contamination, entrained in the air stream as it leaves the spray dryer chamber. The bulk of the contaminated dust is removed from the air stream in a centrifugal dust collector. The remaining dust is removed from the air stream by the water spray of the Venturi Scrubber. A Cyclone Separator removes the entrained water mist injected by the Venturi Scrubber and the cleaned air is discharged to the atmosphere by the blower which induces the air flow through the system. The stack effluent is sampled and analyzed for radioactivity as a check on the performance of the system. The contaminated dust is packed with the settled sludge for sea disposal. The vapor compression and spray drying process reduces the volume of contaminated liquid by a factor of approximately 275.

3. Filtration

Some water which contains natural uranium in particulate form, rather than in solution (such as uranium dioxide), is satisfactorily decontaminated by mechanical filtration. This water is pumped from one storage tank through the filter and into a second storage tank. The filtered water is sampled and analyzed and, if satisfactory decontamination has been achieved, it is released to the Storm Drain. If the decontamination has not been satisfactory, the filtered water is concentrated by evaporation.

4. Ion-exchange

Water which contains a low concentration of dissolved solids, such as the moderator water used in critioal experiment facilities, is treated for the removal of radioactive contamination by passing it through an ion-exchange resin column at a flow rate of six gallons per minute. Water samples are taken after each cycle and the water is released to the plant effluent if the concentration of radioactivity has been reduced to an acceptable level. A decontamination factor of greater than 100 is obtained with this column when used on this type of water.

C. Release and Control

The treated waste liquids enter Bull Run at the source of this stream. Bull Run joins Thompson Run at a point approximately one mile northeast of the Bettis Site and Thompson Run empties into the Monongahela River within the city of Duquesne, Pennsylvania.

FIGURE 3.

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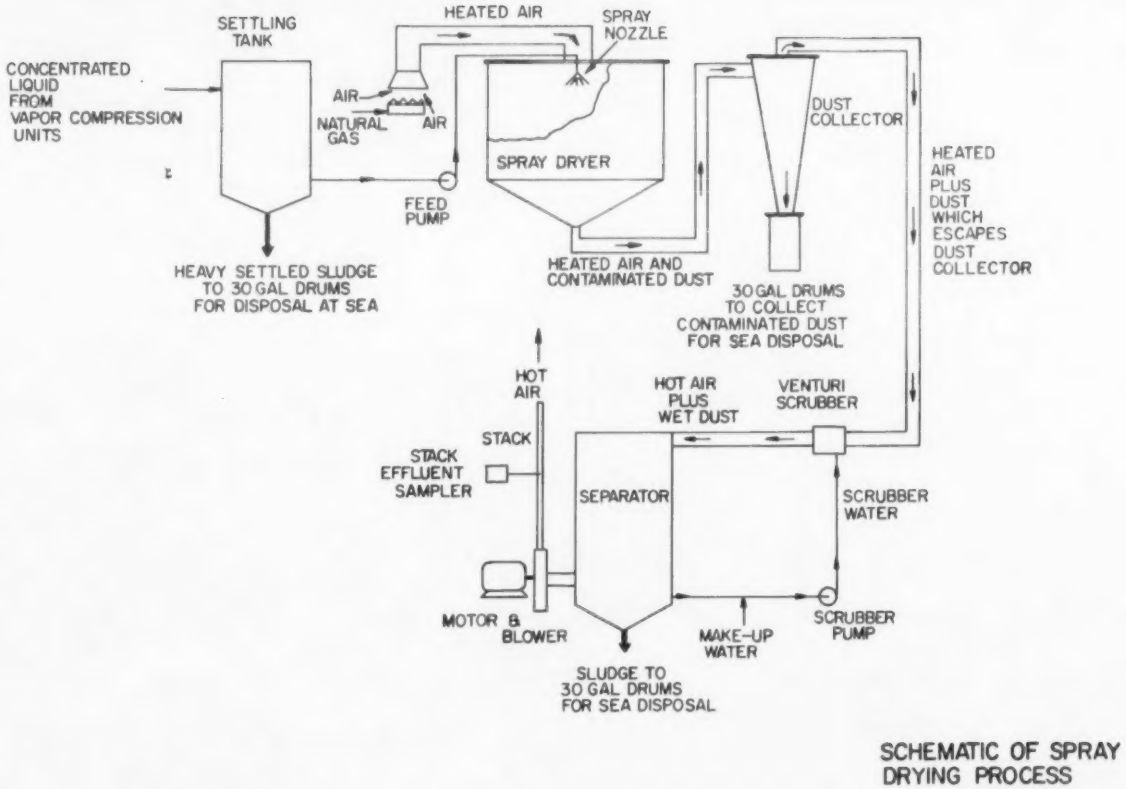
SETTLING TANK
HEATED AIR
SPRAY NOZZLE
CONCENTRATED LIQUID FROM VAPOR COMPRESSION UNITS
AIR
NATURAL GAS
AIR
SPRAY DRYER
DUST COLLECTOR
HEATED AIR PLUS DUST WHICH ESCAPES DUST COLLECTOR
FEED PUMP
HEAVY SETTLED SLUDGE TO 30 GAL DRUMS FOR DISPOSAL AT SEA
HEATED AIR AND CONTAMINATED DUST
30 GAL DRUMS TO COLLECT CONTAMINATED DUST FOR SEA DISPOSAL
HOT AIR
HOT AIR PLUS WET DUST
VENTURI SCRUBBER
STACK
STACK EFFLUENT SAMPLER
SEPARATOR
SCRUBBER WATER
MOTOR & BLOWER
MAKE-UP WATER
SCRUBBER PUMP
SLUDGE TO 30 GAL DRUMS FOR SEA DISPOSAL
SCHEMATIC OF SPRAY DRYING PROCESS

FIGURE 4.

The permissible concentrations for the discharge of radioisotopes in the water effluent from the plant are 7×10^{-6} microcuries of alpha activity per milliliter if there is no beta-gamma activity other than from uranium, and 9×10^{-7} microcuries of beta-gamma activity per milliliter if there is no alpha activity. A fraction of each limit is used in proportion to the concentrations of alpha and beta-gamma radioactivity in the effluent. In actual practice, the radioactivity in the plant effluent is reported as total alpha, beta, gamma microcuries per milliliter as long as this total does not exceed 9×10^{-7} microcuries per milliliter. This alpha limit is one-tenth of the permissible limit for uranium in water as recommended in National Bureau of Standards Handbook 52 and Pennsylvania Regulation 433. With this beta-gamma limit, it has been found that all radionuclides will remain below the permissible limits in National Bureau of Standards Handbook 52 and Pennsylvania Regulation 433.

The average concentration of radioactive materials released each month in the water from the plant site is measured by determining the total radioactivity in the water released from the Monitor and Special Monitor storage tanks and the total water used by the plant as measured by the water meters. The calculated average concentration of radioactive materials (excluding fallout) in the water released from the plant for each month of 1958 is shown in Table 2, Column A. The total radioactivity in the water released from the Monitor and Special Monitor tanks is shown in Table 2, Column C. The average concentration of radioactive materials including the radioactive fallout in the water leaving the plant is measured by the analysis of weekly composite samples collected by the Plant Effluent Sampler (See Figure 5) and the results for each month of 1958 are shown in Table 2, Column B. The Plant Effluent Sampler collects a sample representing an average of all water leaving the main plant area and provides a means of determining the adequacy of the control measures for the release of radioactive materials in the water effluent from the plant.

TABLE 2

Average Concentrations of Radioactivity in the Water Effluent from the Bettis Plant, Bettis Site, and Total Radioactivity Dumped from the Monitor and Special Monitor Tanks during 1958

	AVERAGE CONCENTRATIONS OF RADIOACTIVITY IN WATER EFFLUENT		
	Column A Calculated (Excluding fallout) uc/ml *	Column B Measured (Including fallout) uc/ml *	Column C Total Radioactivity Dumped from Monitor & Special Monitor Tanks Millicuries
January	3.7×10^{-8}	14.4×10^{-8}	2.98
February	6.4×10^{-8}	15.5×10^{-8}	5.15
March	5.7×10^{-8}	11.0×10^{-8}	6.04
April	6.8×10^{-8}	36.5×10^{-8}	6.37
May	10.1×10^{-8}	36.5×10^{-8}	10.98
June	5.3×10^{-8}	20.8×10^{-8}	7.63
July	3.9×10^{-8}	23.9×10^{-8}	4.50
August	4.5×10^{-8}	41.2×10^{-8}	5.16
September	3.9×10^{-8}	29.8×10^{-8}	5.62
October	6.6×10^{-8}	19.3×10^{-8}	7.45
November	3.3×10^{-8} (Est.)	19.4×10^{-8}	3.35
December	4.3×10^{-8} (Est.)	25.2×10^{-8}	4.29

*uc/ml = microcuries per milliliter

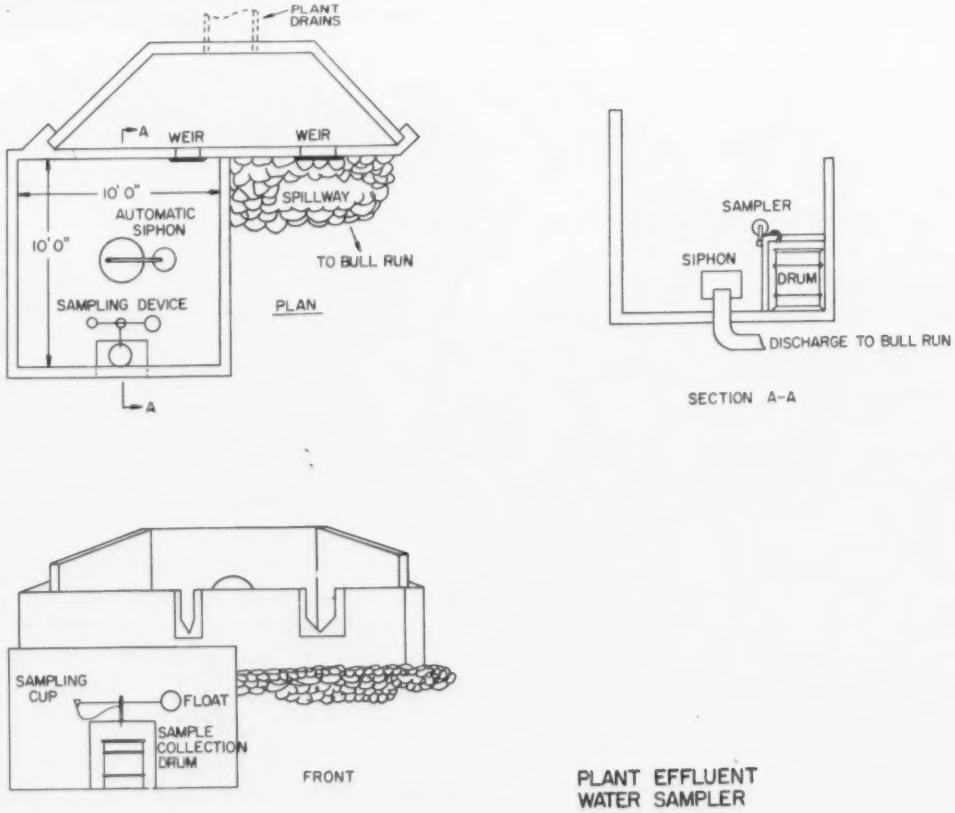
PLANT DRAINS
A
WEIR
WEIR
10' 0"
AUTOMATIC SIPHON
10' 0"
SPILLWAY
TO BULL RUN
SAMPLING DEVICE
PLAN
A
SAMPLER
SIPHON
DRUM
DISCHARGE TO BULL RUN
SECTION A-A
SAMPLING CUP
FLOAT
SAMPLE COLLECTION DRUM
FRONT
PLANT EFFLUENT WATER SAMPLER

FIGURE 5.

III. SOLID WASTES

Radioactive solid wastes are divided into two categories, non-combustibles and combustibles. Radioactive non-combustibles consist of materials such as contaminated glass and metal or irradiated specimens. Combustible wastes consist of any material whose volume can readily be reduced to an ash by incineration. Materials included in this group are paper, wood, cardboard, packing materials and rags.

Non-combustibles are classified as high-level or low-level wastes. Low-level wastes are collected in 30-gallon steel drums located in designated areas near the laboratories or shops which generate these wastes. These drums are periodically collected and transported on pallets by a fork-lift to the Hot Waste Processing Area. In the Hot Waste Area the low-level waste is hand-sorted to insure that each 30-gallon drum is filled to capacity. The 30-gallon drum of waste is then centered inside a 55-gallon drum and the space between the two drums is filled with concrete. The lid is attached to the drum and it is weighed, monitored for radiation intensity, and labeled according to ICC shipping regulations. The 30-gallon drums of waste are sealed in concrete inside 55-gallon drums to insure that the package will weigh at least 750 pounds, so it will sink, and to insure that the package is strong enough to withstand the rigors of transportation to the sea.

High-level non-combustibles are collected from the laboratories in special containers with 1/2" to 6" of lead shielding, capable of holding 1-quart, 2-quart, 1-gallon and 5-gallon cans. This shielding is necessary to reduce the radiation intensity of the high-level wastes to established limits. The containers are monitored to insure that the lead shielding has been fully utilized. They are then centered inside a 55-gallon drum and the drum is filled with concrete. The drum is then processed in the same manner as a drum of low-level waste (see **fig. 6**).

In some cases, lead shielding is not required and a 55-gallon drum with a precast concrete well is used to shield the waste (see **fig. 7**). Two 1-gallon cans of waste are placed in this well and all voids filled with concrete.

Combustible wastes are collected from the laboratories in 55-gallon drums and are transported to the incinerator in the Hot Waste Processing Area. The contents of each drum are checked to insure that no significant quantity of flammable solvent is present which might present a potential hazard during incineration, and to insure that no highly radioactive materials have been placed in the drum. After the operators are sure that the material is safe to incinerate, it is dumped into the top door of the incinerator.

When the incinerator is loaded to its capacity of four 55-gallon drums, the wet dust collector (Rotoclone) system, which removes particulate matter from the incinerator exhaust gases, is placed in operation and

FIGURE 6. Waste shield centered inside a 55-gallon drum.

FIGURE 7. A precast concrete well in a 55-gallon drum.

the charge is ignited. The volume of combustible waste is reduced by a ratio of about 40:1. Ash from the incinerator is collected in a 30-gallon steel drum which is processed in the same manner as low-level non-combustible wastes.

All drums are shipped by common carrier to the U. S. Naval Ammunition Depot, Earle, New Jersey, for disposal by the Navy at AEC approved burial grounds in the Atlantic Ocean in accordance with AEC approved procedures.

All drums must conform with the following requirements. The outside container must be a 55-gallon steel drum with a minimum weight of 750 pounds when filled, to insure rapid sinking in the ocean. Radiation readings on contact with each drum must be less than 175 milliroentgens per hour and less than 10 milliroentgens per hour at one meter from the source.

Reconditioned steel drums are used for the final package and the entire packaging and shipping process has been approved by the Interstate Commerce Commission under the Bureau of Explosives Permit #443.

FIGURE 7. A precast concrete well in a 55-gallon drum.

IV. AIRBORNE WASTES

No significant quantity of radioactive material is discharged to the atmosphere. The airborne wastes consist primarily of the effluent air from dust and fume collectors. These dust and fume collectors are an integral part of the exhaust ventilation systems on processes and laboratory operations which generate airborne radioactive contamination and chemical fumes, dust and vapors. As a general rule, wet collectors are used for processes involving low-level radioactive material. In areas of high-level radioactivity, similar wet collectors are used as precleaners for electrostatic precipitators and/or high efficiency filters. Various dry filter media are used for the collection of dry radioactive particulates. The media is mounted in specially constructed filter housings, or, in some instances, commercially available units such as the Hersey reverse-jet type bag collectors are used.

The stack effluent air is periodically or continuously sampled, depending upon the nature of the operation and the potential for release of radioactive materials to the atmosphere. These samples are analyzed for radioactivity to provide data on the radioactivity released and a check on the performance of the dust and fume collectors.

Approximately 240,000 cubic feet per minute of air is exhausted from the various shops and labs in the plant to remove airborne radioactive contamination and chemical fumes, dusts and vapors from the work areas. Approximately 175,000 cubic feet per minute of this exhaust air comes from processes involving radioactivity, and the remainder (65,000 cubic feet per minute) from processes involving only chemical contaminants.

Permissible levels for the discharge of radioactivity to the atmosphere are established for each plant operation and are based on the type of radioactive contaminant involved. The limits for this release are based on permissible limits established in the National Bureau of Standards Handbook #52. The amount of material discharged to the atmosphere is controlled so that the concentration will not exceed one-tenth of that permitted in National Bureau of Standards Handbook #52.

V. ENVIRONMENTAL SAMPLES

An environmental monitoring program is employed as a check on the adequacy of the control measures which govern the release of radioactive materials from the plant.

Soil samples are collected at designated locations within a half-mile radius of the plant. These samples are analyzed for gross alpha and gross beta activity to determine if there is an increase in the background radiation as a result of the plant operations. Excluding the results from one location within the plant exclusion area where localized radioactive contamination was found, there has been no detectable change in the average gross alpha activity in the soil since the start of plant operations in 1949. The average gross beta activity of the soil has increased from

$(1.4 \pm 0.2) \times 10^{-5}$ microcuries per gram of soil in 1949 to $(2.6 \pm 0.1) \times 10^{-5}$ microcuries per gram of soil in 1958. This increase in the average gross beta activity of the soil is probably due to the accumulation of radioactive fallout from nuclear weapon tests.

Samples of radioactive fallout are collected routinely at eight locations on the plant property. An adhesive-coated paper is exposed in a horizontal position at each sampling location for a period of one week and is then analysed for gross alpha activity and gross beta-gamma activity. The average and the range of the gross beta-gamma activity for samples collected at locations approximating the plant perimeter for each quarter of 1958 are shown in Table 3. There was no detectable alpha activity on the samples collected at these locations.

TABLE 3

Fallout - 1958

	Average Beta-Gamma Activity $uc/ft^2/day$ *	Range of Beta-Gamma Activity $uc/ft^2/day$ *	
		Min.	Max.
1st Quarter	1.75×10^{-4}	3.58×10^{-6}	2.04×10^{-3}
2nd Quarter	5.09×10^{-4}	3.14×10^{-5}	2.31×10^{-3}
3rd Quarter	2.16×10^{-4}	3.72×10^{-5}	6.0×10^{-4}
4th Quarter	3.54×10^{-4}	1.0×10^{-5}	2.26×10^{-3}

* $uc/ft^2/day$ = microcuries per square foot per day

An air monitoring station is operated continuously on the perimeter of the plant and provides data on normal background radiation intensities. During 1958 the average intensity of beta-gamma radiation measured by this station was .019 milliroentgens per hour, and the average intensity of gamma radiation was measured at .0157 milliroentgens per hour. These average intensities do not differ significantly from those obtained by similar monitoring stations located on the PWR Site at Shippingport, Pa.

VI. WASTE DISPOSAL COST AND QUANTITIES

A. Capital Equipment - The Bettis Plant has a capital investment, excluding buildings, of $30,000 in equipment used to process solid radioactive wastes. An investment of $232,000, excluding buildings, has been made for liquid waste processing equipment.

B. Operating Costs - A total of 10,990,000 gallons of waste liquids were collected in the Monitor Tank Rooms during 1958. Of this volume 3,425,750 gallons were processed by evaporation and the remainder, including 356,000 gallons of filtered liquids, were released to the plant effluent (see figure 8). During the same period 4,640 fifty-five gallon drums of packaged solid wastes were shipped for sea burial (see figure 9). This included 570 drums of concentrated liquid wastes.

Table 4 presents the cost of processing this volume of waste during calendar year 1958. The costs presented do not include overhead charges on direct labor, buildings or equipment, nor the Navy charges for burying the wastes in the ocean. All of the December operating costs were not available for this report and in some cases an estimated expenditure, based on the previous 12 months of experience, was used.

C. As each costing system is somewhat unique, a definition of the accounts used in table 4 to present the Bettis Plant waste disposal costs are presented for purposes of clarity.

1. Solid Wastes

a. Direct Labor - This is the labor cost of collecting and processing radioactive solid waste by Industrial Hygiene personnel. It does not include indirect labor costs such as vacations, holidays and other so-called "fringe benefits."

b. Material (Shielding) - High-level radioactive wastes are shielded with lead. This account represents the cost of these shielding materials. It has been separated from the consumable materials account because it represents such a large percentage of the total expenditures.

c. Materials (Consumable) - These are expenditures for consumable type of items such as concrete, 30 and 55-gallon drums, protective clothing, etc.

d. Materials (Maintenance) - This account represents the cost of repair parts needed to maintain the equipment used to process solid wastes. The principle expenditures are repair parts for the incinerator and the dust collector (fume collector).

NUMBER OF GALLONS IN THOUSANDS

RADIOACTIVE WASTE WATER
PROCESSED IN THE MONITOR TANK ROOMS

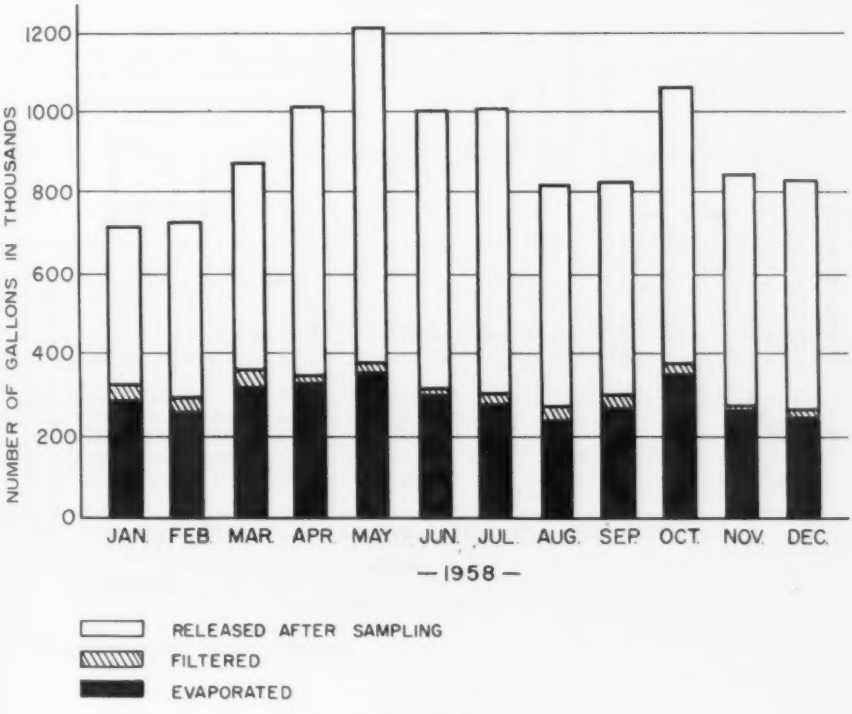

FIGURE 8.

NUMBER OF 55 GALLON DRUMS OF PROCESSED
RADIOACTIVE SOLID WASTES SHIPPED FOR OCEAN BURIAL

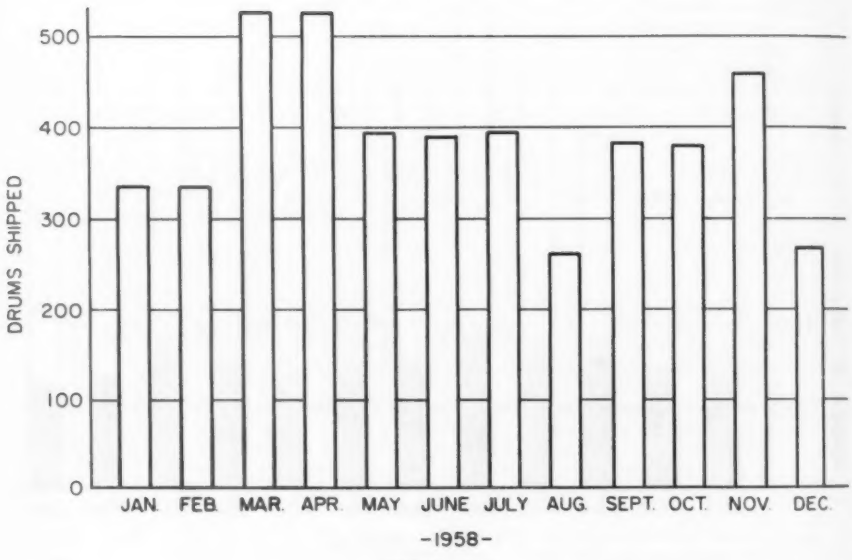

FIGURE 9.

TABLE 4

BETTIS PLANT RADIOACTIVE WASTE DISPOSAL

OPERATING COSTS FOR CALENDAR YEAR 1958

SOLID WASTES

Type Costs	Direct Costs	No. Drums Shipped	Unit Cost Per Drum
1. Direct Labor	$ 11,050	4640	$ 2.38
2. Materials (Shielding)	98,828	"	21.30
3. Materials (Consumable)	41,342*	"	8.91
4. Maintenance Materials	390*	"	.084
5. Maintenance Labor	2,265	"	.49
6. Shipping	25,798*	"	5.56
TOTALS	$179,673	"	$38.72

LIQUID WASTES

Type Costs	Direct Costs	No. Gallons Evaporated	Unit Cost Per Gal. Evap.
1. Direct Labor	$ 58,061	3,425,750	$ .0169
2. Materials (Consumable)	9,249*	"	.0027
3. Maintenance Materials	51,386*	"	.015
4. Maintenance Labor	28,809	"	.0084
5. Fuel Costs	39,214	"	.0114
TOTALS	$186,719	"	$ .0544
GRAND TOTAL	$366,392		

* December costs were estimated.

e. Maintenance Labor - This is the cost of labor from the Works Engineering Department (Plant Maintenance Group) which is needed to keep the processing equipment in good operating condition.

f. Shipping - This is the cost of transporting the processed wastes from the Bettis Plant to the Navy in Earle, N. J.

2. Liquid Wastes

a. Direct Labor - This is the Industrial Hygiene Department labor cost to process liquid wastes. It does not include indirect labor costs such as vacations, holidays and other so-called "fringe benefits."

b. Materials (Consumable) - These are expenditures for consumable type of items such as absorbant material used to thicken the concentrated wastes, sample containers, neutralizing materials, etc.

c. Maintenance Materials - This is the cost of repair parts purchased to maintain the processing equipment (boilers, evaporators, pumps, etc.) in good operating condition.

d. Maintenance Labor - This is the cost of labor from the Works Engineering Department (Plant Maintenance Group) which is needed to keep the processing equipment in good operating condition.

e. Fuel Costs - The means for determining the amount of gas and electricity used in processing these wastes will not be available for some time yet. As an interim measure, costs have been estimated using the design capacities of the equipment as the basis for obtaining a "cost per gallon evaporated figure."

VII. SUMMARY

At Bettis Plant all radioactive liquid wastes are collected in special drain systems to prevent any accidental release to the Monongahela River. Liquids containing small amounts of radioactivity, within prescribed limits, are released and the balance is concentrated. The liquid concentrates are ultimately disposed of in the Atlantic Ocean as are all solid radioactive wastes. Air from processes involving radioactive materials is passed through various types of collectors, depending on the process, which reduce the concentration of such materials in the air to safe levels prior to discharging it to the atmosphere. A continuous site monitoring program involving the collection of fallout and soil samples is conducted to provide a check on radioactivity in the plant environs.

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RADIOACTIVE WASTE DISPOSAL FROM U. S. NAVAL NUCLEAR POWERED SHIPS

Prepared by: T. J. Iltis and M. E. Miles, Nuclear Propulsion Divisions, Bureau of Ships, Department of the Navy

I SUMMARY

The purpose of this report is to: (1) describe the sources and nature of radioactive wastes from U. S. Naval nuclear powered ships, (2) outline and discuss the established waste disposal procedures used for these ships, and (3) summarize the measurements made to detect any effect of wastes discharged by the first ships on the radioactivity of their harbor environs.

The basic criterion adopted for disposal of radioactive waste from U. S. Naval nuclear powered ships is that disposal should not increase the average concentrations of radionuclides in the surrounding environment by more than one-tenth of the maximum permissible concentrations for continuous exposure listed in National Bureau of Standards Handbook 52. Actual data from the operating ships shows that the radioactivity of their wastes is consistently low and has had no detectable effect on the radioactivity of their environment.

II INTRODUCTION

A. General Description of U. S. Naval Nuclear Power Plants.

All nuclear propelled U. S. Naval Ships now planned, in construction, or in operation are powered by pressurized water reactors. In these reactors, pressurized water circulating through the reactor core picks up the heat of the nuclear reaction. The reactor coolant passes through heat exchangers which transfer the heat to water in a steam system. The steam system water is converted to steam and is then used as the source of power for the propulsion plant as well as for the auxiliary machinery.

B. Principal Sources of Radioactive Waste.

The principal source of radioactive waste from all nuclear powered naval ships is the reactor coolant water which contains small quantities of activated impurities. The largest amounts of reactor coolant water are discharged when this water expands as a result of bringing the reactor plant up to operating temperature. This normally happens a few times per month on each ship and the quantity of coolant water discharged on each heat-up averages about 500 gallons. The nature of the radioactivity in the coolant water and the procedures established for its disposal are described in detail in the main body of this report (Section III).

There are other sources of radioactive waste derived from the operation of naval nuclear power plants which require only infrequent waste disposal considerations. These include disposal of (a) the ion exchange resin that is used to purify the coolant water of the reactor plant, (b) reactor shield water, (c) solid wastes from maintenance operations, and (d) special wastes

from laundry or decontamination operations. The nature and procedures for handling these wastes from infrequent operations are discussed in Section IV of this report.

Aside from the above mentioned sources of radioactive waste, it should be noted that the direct radiation emanating from the hull of any nuclear powered naval ship is designed to be insufficient to cause any detectable activation of the sea water.

C. Fission Products.

In addition to activated impurities, the reactor cooling water may contain trace amounts of fission products. These come from minute quantities of uranium impurity in the reactor structural materials. The large quantities of fission products that are produced by the fission process of the reactor are retained where they are born: metallurgically bound within the fuel alloy. They cannot get out of the fuel elements unless by a very remote possibility the reactor itself were to melt down. The plant is protected against this possibility by its inherent safety characteristics and by a "fail-safe" protection system, although the possibility of such an accident cannot be completely eliminated. If the ship were sunk, it is expected that the reactor core could remain submerged in sea water for decades without release of fission products, since the zirconium protective cladding on the fuel elements corrodes only a few millionths of an inch per year. The steam system is entirely separate from the reactor system, and a failure in the steam system would not endanger the reactor nor release fission products or activated impurities.

D. Environmental Effects.

In order to confirm the adequacy of waste disposal procedures for naval nuclear powered ships, surveys are being conducted of the radioactivity in the environment around ports where nuclear ships are being built and operated. These surveys cover periods both before and after commencement of operation of the reactors. A description of these surveys and some results obtained to date are discussed in Section V of this report.

III Reactor Coolant Waste

A. Nature of Reactor Coolant.

In order to establish waste disposal procedures, the radioactive nuclides in the reactor coolant must be identified. The radioactive isotopes of most concern result from activation of the small amounts of corrosion and wear products of plant surfaces and from activation of small amounts of impurities in the coolant water. All nuclides normally present in significant amounts in the reactor coolant of operating naval reactors and prototypes have been identified and their concentrations determined. These determinations are performed during initial operations of the reactors, and at intervals thereafter, coolant samples are analyzed for the long lived (greater than several days half life) nuclides. These analyses have shown that the concentrations of radioactive nuclides in reactor coolant a few minutes after sampling for naval nuclear propulsion plants are approximately as shown in Table I. The table shows for comparison the tolerances used for dumping reactor coolant in restricted waters, to be discussed in Section III B of this paper. Significant increases in coolant activity would be indications of reactor plant malfunction; therefore, continued reactor operation with the concentration of any nuclide above dumping tolerance is not expected and would not normally be permitted.

Table I

Concentrations of Radionuclides in Reactor Coolant

Nuclide	Half Life	Measured Activities of Coolant in microcuries per cubic centimeter		Dumping Tolerance in μc/cc
		Maximum	Average	
Mn^{56}	2.5 hours	9.3×10^{-2}	2.2×10^{-2}	15
Co^{60}	5.2 years	2.5×10^{-2}	5.7×10^{-6}	2
Fe^{59}	45 days	2.8×10^{-3}	1.5×10^{-4}	1×10^{-2}
Ni^{65}	2.56 hours	1.3×10^{-3}	1.6×10^{-4}	1.9
Cr^{51}	27 days	5.5×10^{-3}	1.0×10^{-5}	'50
Na^{24}	15 hours	2.0×10^{-2}	8.0×10^{-5}	3×10^{-1}
Cu^{64}	12.8 hours	9.1×10^{-3}	1.5×10^{-5}	8
Ta^{182}	112 days	5.6×10^{-2}	7.3×10^{-3}	10
F^{18}	1.87 hours	6.8×10^{-2}	1.2×10^{-2}	90
W^{187}	24 hours	9.0×10^{-3}	3.3×10^{-4}	9×10^{-2}
Gross Activity Measured 15 min after Sampling		1.5×10^{-1}	5.0×10^{-2}	3.0
Gross Activity Measured 120 hours after Sampling		3.6×10^{-2}	3.1×10^{-3}	1×10^{-1}

Complete results of the nuclide identification program performed on NAUTILUS and SKATE coolant water are shown in Appendix B. Detailed analyses of NAUTILUS reactor coolant were made every 1 to 2 months during the operation of the first core and at less frequent intervals for the second core. No new nuclides appeared in these analyses. The gross 15 minute activity has remained below a normal operating limit of 0.3 microcuries per cubic centimeter (μc/cc). Similar results have been obtained for the SKATE and other naval reactors already operating. Nevertheless, initial and periodic checks on the activity of the nuclides present in the reactor coolant of each new ship will continue to be made to detect changes that may affect plant operation or waste disposal procedures.

Fission products occur in very small concentrations in reactor coolant because of uranium impurity in core structural materials. The concentrations of fission products in reactor coolant from this source are shown in Table II, along with their tolerances for waste disposal. (The very short lived fission products are not shown in the table since their tolerances are relativsly high.) Since a rise in fission product concentrations would indicate a fuel element defect that might lead to further plant operational or maintenance difficulties, continued operation of a naval reactor with fission products concentrations above waste disposal limits would not normally be permitted.

Table II

Concentrations of Fission Products in Reactor Coolant

Nuclide	Half Life	Concentration, μc/cc	Dumping Tolerance, μc/cc
total iodine	variable	5×10^{-4}	to be conservative 1×10^{-3} is normally used
I^{131}	8 days	1×10^{-5}	3×10^{-3}
total strontium	variable	5×10^{-5}	to be conservative 1×10^{-4} is normally used
Sr^{90}	28 years	5×10^{-8}	8×10^{-5}
Sr^{89}	54 days	5×10^{-6}	7×10^{-3}
Ba^{140}	13 days	1×10^{-6}	2×10^{-1}
Ce^{144}	285 days	1×10^{-7}	4.0
Cs^{137}	30 years	1×10^{-8}	1.5×10^{-1}

Two other radionuclides that may be present in reactor coolant but produce no waste disposal problems are argon-41 and tritium. Argon, present as a constituent in small quantities of air dissolved in reactor coolant, becomes activated as argon-41 and may reach concentrations up to 0.7 μc/cc. Argon-41 has a short half-life (1.9 hours); it is gaseous and thus it will mostly escape to the atmosphere upon discharge. It is not an ingestion hazard, and calculations have shown that if all the argon-41 in the coolant were released as a cloud, persons exposed to the cloud would receive only a negligible external radiation dose. Since argon-41 is no waste disposal problem, it will not be further discussed in this report.

In some of the reactors the primary coolant water is treated with a few parts per million of lithium hydroxide to raise the pH (alkalinity) of the coolant and thereby reduce corrosion of the system materials. The lithium will undergo a neutron reaction in the reactor core and form small amounts

of tritium in the coolant. Experience with the PWR plant at Shippingport, Pa., which uses this same lithium hydroxide treatment, has shown that the tritium activity in primary coolant will consistently remain below 0.3 μc/cc. This concentration of tritium activity is far less than dumping tolerance of 7 μc/cc established for tritium and thus it does not constitute a waste disposal problem.

B. Criteria and Procedures for Disposal of Reactor Coolant.

The basic criterion adopted for disposal of coolant is that disposal should not increase the average concentrations of radionuclides in the surrounding environment by more than one-tenth of the maximum permissible concentrations for continuous exposure listed in National Bureau of Standards (NBS) Handbook 52. Application of this criterion has led to the waste disposal instruction for use by all U. S. Naval nuclear-powered ships, included as Appendix A to this report. This instruction was reviewed and concurred in by the U. S. Public Health Service, the Bureau of Medicine and Surgery of the Navy, the Reactor Development and Biology and Medicine Divisions of the AEC, and the Atomic Energy Applications Division of the Office of the Chief of Naval Operations. The instruction states that for discharge in port, reactor coolant gross activity must be less than 3 μc/cc and the fission product iodine-131 must be below 10^{-3} μc/cc. For discharge in the open sea, the instruction specifies no restriction because the quantities of waste are small, the activities are normally well below the above dumping tolerances, and the dilution obtained in the open sea is very much greater than in ports.

The development of simple shipboard control procedures from the basic criterion stated above is discussed below in summary and then in detail:

1. A dilution factor is first determined by considering the frequency and quantities of coolant that are discharged and the dilution factors that are available in harbor waters. It has been established that in these waters a dumping tolerance of 100 times the NBS Handbook 52 tolerance for any specific nuclide is conservative.
2. By equally scaling up the activities of all the individual nuclides until the first nuclide reaches its dumping tolerance, it is determined that when the gross coolant activity is 3 μc/cc all nuclides are below their respective dumping tolerances. This gross activity then becomes the shipboard limit for disposal of reactor coolant during operation.
3. A similar scaleup is performed for the coolant after reactor shutdown. This condition is different from the operating condition above since all short lived nuclides in the reactor coolant have decayed. The gross activity limit becomes 0.1 μc/cc for this case.
4. For the unexpected condition where fission product concentrations might become significant, the fission product that is closest to dumping tolerance is determined. A shipboard radiochemical procedure is established for determination of this nuclide. Experience on all naval ships to date indicates that this worst nuclide is iodine^{131} with a dumping tolerance of 3×10^{-3} μc/cc.

Each of the above steps is explained in more detail as follows:

1. Dilution factor and definition of dumping tolerance.

Because coolant is normally discharged in small quantities (average less than 500 gallons per discharge) it has been assumed that the discharged water will almost immediately be diluted in the harbor water by a factor of at least 1000*. This assumption has been checked by actual measurements of activity in the water alongside the NAUTILUS while the ship was discharging reactor coolant at the Electric Boat Division dock in Groton, Connecticut. These measurements showed that when the ship was discharging water at about 2×10^{-2} μc/cc the harbor water alongside the ship remained at background level of about 2×10^{-7} μc/cc, representing a dilution factor of at least 100,000.

To be conservative, a dilution factor of only 1000 has been assumed. With this factor, dumping tolerances in the Navy instruction (Appendix A) are set at 100 times the maximum permissible concentrations for continuous exposure listed in NBS Handbook 52. This will insure that disposal will meet the basic criterion of not increasing the average concentrations of radionuclides in the environmental water by more than one-tenth of the NBS Handbook 52 concentrations. The dumping tolerances in this report and in Appendix A then are defined as 100 times the concentrations listed in NBS Handbook 52.

* A volume of water equal to the displacement of the ship would be more than enough to accomplish this dilution.

2. Gross activity limit during reactor operation.

As noted previously a complete analysis of all the significant radionuclides in the reactor coolant is made for each naval ship. (See Table I and Appendix B) If these concentrations of nuclides in the reactor coolant are scaled up for a gross coolant activity of 3 μc/cc, one may observe (see Table III) that the concentrations of all nuclides are below dumping tolerance. On this basis, a shipboard procedure is used to verify that gross activity is below 3 μc/cc. This procedure is easy to perform by ship personnel and provides ready control for coolant waste disposal purposes. In actual experience the gross activity is nearly always below 0.3 μc/cc. Significant increases of coolant activity above this level would be indications of plant malfunction and continued reactor operation would normally not be permitted. Hence the normal limit imposed for reactor operating reasons is one-tenth the waste disposal limit of 3 μc/cc.

Table III

Comparison of Nuclide Activities with Dumping Tolerance
(all activities in μc/cc)

Nuclide	Column A Nuclide Activity from Table I scaled up to gross activity = 3 μc/cc	Column B Dumping Tolerance (100 times Handbook 52)	Ratio Column A Column B
Mn^{56}	1.3	15.0	.09
Co^{60}	3.6×10^{-4}	2.0	.002
Fe^{59}	9.0×10^{-3}	1.0×10^{-2}	.90
Ni^{65}	9.6×10^{-3}	19.0	.0005
Cr^{51}	6.0×10^{-4}	50.0	.00001
Na^{24}	4.8×10^{-3}	8.0×10^{-1}	.006
Cu^{64}	9.0×10^{-4}	8.0	.0001
Ta^{182}	4.3×10^{-1}	10.0	.04
F^{18}	7.2×10^{-1}	90.0	.008
W^{187}	1.9×10^{-2}	9.0×10^{-2}	.21

3. Gross activity limit after reactor shutdown.

Within several days after reactor shutdown there are no longer significant quantities of short-lived nuclides; the coolant activity is now largely composed of long-lived isotopes such as iron 59 and cobalt 60. A different gross activity limit is needed for waste disposal after shutdown, since if the water had a gross activity as high as 3 μc/cc, some of these nuclides could be above their dumping tolerances. Examination of Table I and Appendix B shows that iron 59 has concentrations closest to dumping tolerance and that its highest concentration in the coolant more than 48 hours after shutdown is less than 10 per cent of the gross activity. On this basis, if the gross activity is less than 0.1 μc/cc 48 hours or more after shutdown, all nuclides including iron 59 will be below dumping tolerances. Table IV illustrates this point. On the basis of these data the instruction to ships, Appendix A, uses the limit of 0.1 μc/cc for times beyond 48 hours after reactor shutdown.

Table IV

Reactor Coolant Activity 48 Hours After Shutdown

(Scaled up to a gross activity of 0.1 μc/cc assuming the most adverse Fe^{59} concentrations)

Nuclide	Half Life	Measured Coolant Activity (μc/cc)	Dumping Tolerance (μc/cc)
Fe^{59}	45 days	1×10^{-2}	1×10^{-2}
Co^{60}	5.2 years	1.2×10^{-2}	2
Cr^{51}	27 days	2.6×10^{-3}	50
Na^{24}	15 hours	3.2×10^{-2}	8×10^{-1}
Cu^{64}	12.8 hours	1.5×10^{-2}	8.0
Ta^{182}	112 days	2.6×10^{-2}	10
W^{187}	24 hours	1.5×10^{-3}	9×10^{-2}
gross activity		1.0×10^{-1}	1×10^{-1}

4. Fission products:

Analysis of the fission product concentrations in the coolant (shown in Table II) shows that iodine 131 comes the closest to its dumping tolerance. The relative amounts of all these fission products has remained consistent on all naval reactor plants operated to date. For shipboard control of coolant waste disposal then, a radiochemical analysis for iodine is performed daily to determine that iodine 131 concentration is below 10^{-3} μc/cc. This will ensure that all other fission products are below their respective dumping tolerances.

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IV WASTES FROM INFREQUENT OPERATIONS

Reactor plant operations also produce other radioactive wastes that require less frequent disposal than reactor coolant. This section discusses these wastes which include ion exchange resin, reactor shield water, solid wastes from maintenance operations, and special wastes from laundry or decontamination operations.

A. Ion Exchange Resin.

The reactor coolant is continuously being filtered and purified by passing a small bypass flow through an ion exchange resin. This resin becomes exhausted and is replaced approximately every six months. Table V shows the radioactivity associated with the spent resin. Short lived nuclides are not included since the quantities present shortly after reactor shutdown are insignificant compared to the long-lived nuclides shown.

Table V

Radioactivity of Spent Ion Exchange Resin*

Nuclide	Half Life	Maximum Activity,* curies
Co^{60}	5.2 years	10
Co^{58}	71 days	0.5
Fe^{59}	45 days	0.5
Cr^{51}	27 days	0.3
Mn^{54}	300 days	0.2
Hf^{175}	70 days	1
Total		12.5

* Maximum radioactivity expected based on measurements from operating plants.

If resin replacement is necessary in port, the resin is dumped to a disposable catch tank. The catch tank is subsequently sealed and buried by land or sea in accordance with approved procedures.

Resin discharge at sea can take account of the great dilution available in the ocean. When dumped overboard the resin will sink and as it sinks the radioactive ions on the resin are rapidly replaced by ions of the sea water. Thus, within a few minutes the radioactivity has transferred from the resin to the sea water in the wake of the ship where it will readily disperse. Assuming conservatively that the wake is no larger than the path of the ship itself, the distributed activity from the resin results in a sea water gross concentration in the ship's wake of less than 10^{-3} μc/cc. Even at this concentration all nuclides are below NBS Handbook 52 permissible concentrations. In addition, subsequent action of wind, wave, and current will rapidly decrease these concentrations. On this basis the navy instruction Appendix A allows resin disposal in the ocean. However, in order to avoid any possibility of having such discharges increase the radioactivity to which people are exposed, restrictions stated in Appendix A are placed on ship location with respect to land, to other ships, and to fishing areas during resin discharge.

B. Reactor Shield Water.

Some attenuation of the radiation emanating from the reactor core is accomplished by using water in a shield tank around the reactor. This shield water will seldom if ever be dumped during the life of a ship. Two-tenths per cent potassium chromate is used in this water as a corrosion inhibitor to protect

the steel surfaces inside the tank. Neutron activation of the potassium chromate and impurities in the shield water produce small concentrations of radionuclides as shown in Table VI. Since the concentrations are well below dumping tolerance, no waste disposal restrictions for radioactivity are necessary on disposing of shield tank water.

Table VI

Radionuclides in Shield Tank Water

Nuclide	Maximum Observed Activity, μc/cc	Dumping Tolerance, μc/cc
K^{42}	0.2	1
Cr^{51}	0.5	5
Fe^{59}	1×10^{-4}	1×10^{-2}

C. Solid Wastes.

Solid radioactive wastes from nuclear ships result primarily from maintenance operations. Such materials include metal scrap, pieces of insulation, rags, sheet plastic, and paper. These solid wastes are given by the ships to shore or tender facilities for subsequent packaging and burial in accordance with approved procedures.

D. Decontamination and Laundry Wastes.

Two other operations associated with reactor plants require disposal of radioactive liquids. Decontamination of radioactive tools and equipment and laundering of radioactive anti-contamination clothing may be performed on some ships. Discharge of resulting liquids is permitted by Appendix A in harbors if no nuclide concentrations exceed dumping tolerance. To insure that this criterion

is met the radioactive decontamination and laundry wastes from the ships are held up for monitoring and treatment by ion exchange if necessary.

It should be noted that all of the above wastes from infrequent operations with the exception of shield water derive their contamination from radioactivity produced in the reactor coolant water. They will therefore contain the same radionuclides as the reactor coolant.

V ENVIRONMENTAL EFFECTS

In order to verify the adequacy of the ship waste disposal procedures of Appendix A, surveys are being conducted of the radioactivity in the environment around ports where nuclear ships are being built and operated. These surveys cover periods both before and after commencement of operation of the reactors. The surveys are being made in cooperation with the U. S. Public Health Service and are being conducted by state and local public health organizations and the shipyards. Included in the surveys are measurements of the radioactivity of water and atmospheric fallout, and sometimes marine organisms, fish, shellfish, mud, and vegetation.

The greatest amount of experience with these surveys to date has been obtained in the New London, Connecticut area where the USS NAUTILUS has operated for 4 years and the USS SKATE has operated 1 year. The waste disposal records of these ships are shown in Tables VII and VIII. They are presented for later comparison with environmental survey results, and they may be summarized as follows:

Average gallons dumped in New London Harbor per month - - - - - - - - - - - - - 625

Average number of discharges per month - 7

Average number of gallons dumped in a single discharge- - - - - - - - - - - - 90 gal

Maximum 15m activity at any single discharge - - - - - - - - - - 2.0×10^{-1} μc/cc

Average 15m Activity of Water Dumped - - - - - - - - - - - - - - - 2.0×10^{-2} μc/cc

Average 120 hour Activity of Water Dumped - - - - - - - - - - - - - 4.5×10^{-3} μc/cc

In addition to the above data from NAUTILUS and SKATE on disposal at New London, a review of coolant activity on all operating ships indicates the activity of the reactor coolant water is always low, averaging 2.5×10^{-2} μc/cc, and never reaches the normal limit of 0.3 μc/cc. Moreover, the logs indicate that at no time has water of an activity greater than 0.3 μc/cc been disposed of in harbor or at sea.

Table VII

USS SKATE Reactor Coolant Discharged in New London Harbor

Date	Gallons	Average 15 minute Activity
Dec 1957	485	1.75×10^{-2}
Jan 1958	844	1.47×10^{-2}
Feb 1958	120	1.43×10^{-2}
Mar 1958	--	--
Apr 1958	130	0.90×10^{-2}
May 1958	--	--
Jun 1958	--	--
Jul 1958	806	0.83×10^{-2}
Aug 1958	--	--
Sep 1958	--	--
Oct 1958	578	0.15×10^{-2}
Nov 1958	316	0.83×10^{-2}
Dec 1958	328	0.34×10^{-2}

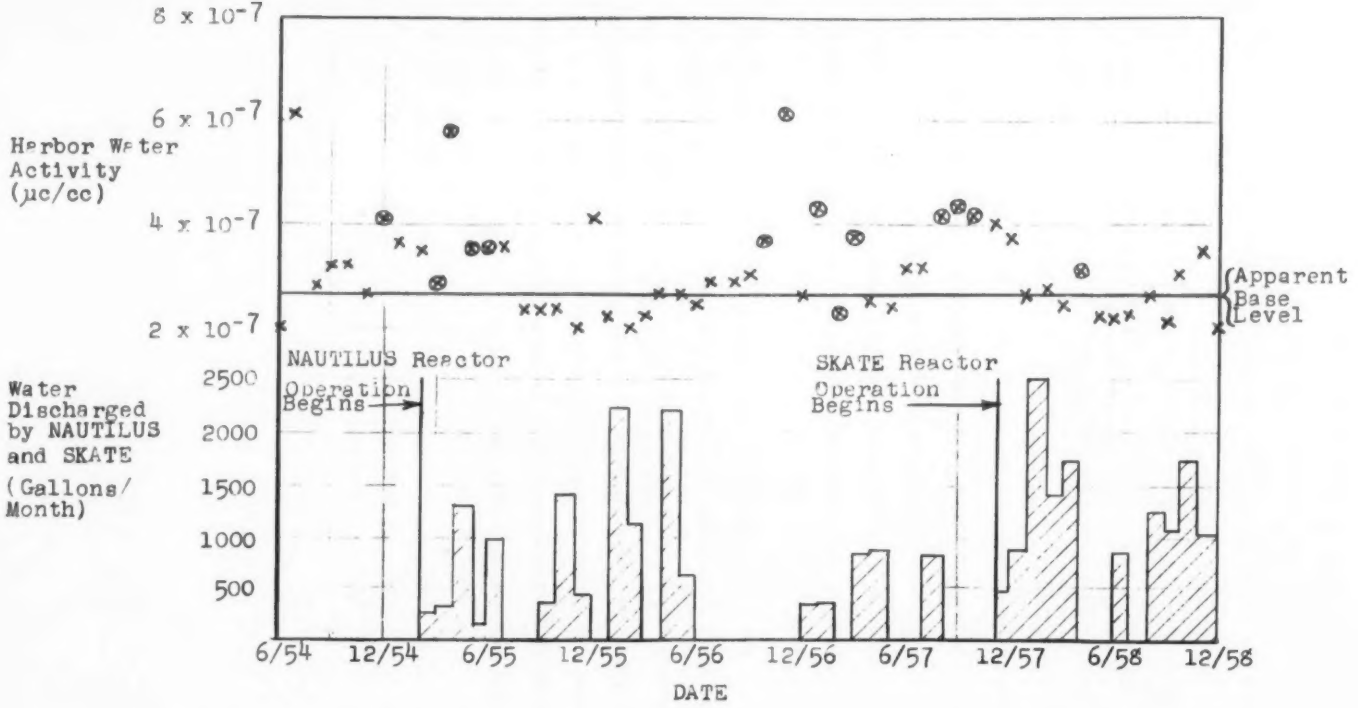
Table VIII

USS NAUTILUS Reactor Coolant Discharged in New London Harbor

DATE	GALLONS	AVERAGE 15 MINUTE ACTIVITY	DATE	GALLONS	AVERAGE 15 MINUTE ACTIVITY
Mar 1955	255	1.26×10^{-2}	Jan 1957	340	1.00×10^{-2}
Apr 1955	300	0.23×10^{-2}	Feb 1957	354	2.00×10^{-2}
May 1955	1357	2.11×10^{-2}	Mar 1957	--	--
Jun 1955	148	0.65×10^{-2}	Apr 1957	800	0.70×10^{-2}
Jul 1955	998	2.41×10^{-2}	May 1957	880	0.67×10^{-2}
Aug 1955	--	--	Jun 1957	--	--
Sep 1955	--	--	Jul 1957	--	--
Oct 1955	366	0.30×10^{-2}	Aug 1957	800	3.36×10^{-2}
Nov 1955	1444	1.29×10^{-2}	Sep 1957	--	--
Dec 1955	417	1.91×10^{-2}	Oct 1957	--	--
Jan 1956	--	--	Nov 1957	--	--
Feb 1956	2268	1.78×10^{-2}	Dec 1957	--	--
Mar 1956	1128	2.97×10^{-2}	Jan 1958	--	--
Apr 1956	--	--	Feb 1958	2380	0.90×10^{-2}
May 1956	2236	1.41×10^{-2}	Mar 1958	1378	3.06×10^{-2}
Jun 1956	612	0.89×10^{-2}	Apr 1958	1608	2.76×10^{-2}
Jul 1956	--	--	May 1958	--	--
Aug 1956	--	--	Jun 1958	--	--
Sep 1956	--	--	Jul 1958	--	--
Oct 1956	--	--	Aug 1958	--	--
Nov 1956	--	--	Sep 1958	1256	1.56×10^{-2}
Dec 1956	--	--	Oct 1958	540	1.41×10^{-2}
			Nov 1958	1428	4.56×10^{-2}
			Dec 1958	696	3.01×10^{-2}

As noted above, a program was initiated before NAUTILUS began operation to survey periodically the activity of the New London Harbor environs. This survey was carried out by the Connecticut Department of Health, the Connecticut water Commission, and the Electric Boat Division. An analysis of the effects of discharging of the reactor plant wastes on the activity of the Thames River can be made by considering the discharges presented in Tables VII and VIII together with the pre- and post-operational survey data obtained by the Electric Boat Division. Fig. 1 presents the record of the NAUTILUS and SKATE discharges at or near the Electric Boat dock together with the results of the operational surveys of water activities in the immediate area of the dock. The data show that the post-operational survey measurements all fall well within the range of the pre-operational measurements even during periods of maximum discharge from the ships. The data further indicates that the only increases in the activity of the river water samples bear no relation to the operation of nuclear submarines, but occurred during periods when nuclear weapons fallout was detected in rain and snow samples.

Examination of the additional survey data taken by the Connecticut Department of Health and The State Water Commission reveals that nearly all points surveyed both pre- and post-operational, showed activity below the minimum detectable limit. Some scattered pre- and post-operational data points show activities ranging up to twice the minimum detectable activity. These points bear no correlation with NAUTILUS discharges or with location respective to the Electric Boat dock where the NAUTILUS discharges were centered.



× Normal survey. Each point represents the maximum of about 60 measurements per month

⊗ Survey point coincides with above-background fallout from nuclear weapons

FIGURE 1. Activity of water in New London harbor near Electric Boat division.

One must conclude from this survey data that the controlled disposal of radioactive wastes from the NAUTILUS and SKATE has had no detectable effect on the radioactivity of the environment. The surveys will continue as more ships go into operation at New London and other ports to detect any possible effects of the naval ships on environmental radioactivity. Surveys are now underway at Portsmouth, Naval Shipyard, New Hampshire; Mare Island Naval Shipyard, California; and Bethlehem Steel Shipbuilding Division, Quincy, Massachusetts. Additional surveys will soon be initiated at Newport News Shipbuilding and Drydock Company, Virginia, and Ingalls Shipbuilding Corporation, Pascagoula, Mississippi.

FIGURE 1. Activity of water in New London harbor near Electric Boat division.

APPENDIX A

DEPARTMENT OF THE NAVY
BUREAU OF SHIPS
WASHINGTON 25, D. C.

IN REPLY REFER TO

BUSHIPS 9890.5
Ser 1500G-628
12 May 1958

BUSHIPS INSTRUCTION 9890.5

From: Chief, Bureau of Ships
To: Distribution List

Subj: Disposal of Radioactive Effluents from U. S. Naval Nuclear-Powered Ships

Encl: (1) Maximum Permissible Concentrations of Radioisotopes in Effluents for Discharge Within 12 Miles from Shore

1. Purpose: To establish procedures and limits for the disposal of radioactive effluents from U. S. naval nuclear-powered ships.

2. Cancellation: This instruction cancels and supersedes the following:

(a) BuShips ltr SSN-571 (590) Ser 590-1808 dtd 31 May 1955
(b) BuShips ltr Ser 1500G-660 dtd 26 September 1957

3. Scope: This instruction applies, during construction or operation, to all U. S. naval nuclear-powered ships equipped with pressurized-water reactor plants.

4. Reactor Coolant Sampling: The following analyses are performed using shipboard type Geiger-Mueller counting equipment for reactor plant operational control, and to obtain information for disposal of reactor coolant:

(a) A sample of reactor coolant is taken at least once daily and measured for short-lived activity at 15 ±2 minutes after sampling. At least once a week the same 15 minute sample is allowed to decay for 120 ±6 hours and measured again for long-lived activity.

(b) The concentrations of fission products in the reactor coolant water are determined by analyses performed daily for iodine activity and monthly for strontium activity.

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12 May 1958

5. Procedures for Disposal of Reactor Coolant Water:

(a) When the ship is more than 12 miles from shore, reactor coolant water and other effluents containing reactor coolant as the only radioactive contaminant may be discharged directly overboard without restriction.

(b) Reactor coolant water and other effluents containing reactor coolant as the only radioactive contaminant may be discharged directly overboard within 12 miles from shore, or at dockside, provided:

(1) the daily iodine analysis and monthly strontium analysis indicate that concentrations of fission products are less than 10^{-3} microcuries per milliliter (μc/ml) of iodine 131 and less than 10^{-4} μc/ml of strontium 90.

(2) when the reactor is in operation or has been shut down for less than 48 hours, the daily measurement of gross degassed activity, 15 minutes after sampling, does not exceed 3 μc/ml.

(3) when the reactor has been shut down for 48 hours or more, the daily measurement of gross degassed activity, 15 minutes after sampling, does not exceed 0.1 μc/ml.

(c) If the conditions in paragraph 5 b above are not met, reactor coolant water should not be discharged overboard within 12 miles from shore. Under these conditions reactor coolant water may be discharged to a dockside retention tank for disposal by a shore facility.

6. Procedures for Disposal of Spent Demineralizer Resin:

(a) At sea, spent demineralizer resin may be discharged overboard when the ship is more than 12 miles from shore, provided:

(1) the ship has headway on,

(2) other ships are not within 3 miles, and

(3) the ship is not in known fishing areas.

(b) Except as provided above, spent demineralizer resin should not be discharged overboard. At dockside, resin may be discharged to a retention tank for disposal by a shore facility.

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7. Procedures for Disposal of Other Radioactive Effluents:

(a) When the ship is more than 12 miles from shore, radioactive effluents derived from other propulsion plant fluid systems, shipboard decontamination, and laundry may be discharged directly overboard without restriction.

(b) For discharge directly overboard within 12 miles from shore or at dockside, the radioactivity of liquid effluents derived from propulsion plant fluid systems, shipboard decontamination, and laundry must be less than the values listed in enclosure (1).

8. Records:

(a) Records shall be maintained of the measurements of paragraph 4 as well as the type of discharge, total quantity, time, and location of discharged radioactive effluents.

(b) After the first six months of operation, the records obtained in accordance with the preceding requirement should be submitted to the Bureau of Ships for review.

9. Effective Date: This instruction is effective upon receipt.

H. G. Rickover
Assistant Chief of Bureau
for Nuclear Propulsion

BUSHIPSINST 9890.5
12 May 1958

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12 May 1958

Enclosure (1)
Maximum Permissible Concentrations of Radioisotopes in Effluents

For Discharge Within 12 Miles From Shore

(Derived From National Bureau of Standards Handbook 52)

Isotope	Concentration (μc/ml)	Isotope	Concentration (μc/ml)
Unidentified beta or gamma emitters or any undetermined mixtures of beta or gamma emitters	1×10^{-5}	Unidentified alpha emitters or any undetermined mixtures of alpha emitters	1×10^{-5}
$Ba^{140} + La^{140}$	2×10^{-1}	Na^{24}	3×10^{-1}
C^{14}	3×10^{-1}	Nb^{95}	4×10^{-1}
$Cd^{109} + Ag^{109}$	7	Ni^{59}	
$Ce^{144} + Pr^{144}$	4	Po^{210}	3×10^{-3}
Cl^{36}	2×10^{-1}	Pu^{239}	1.5×10^{-4}
Co^{60}	2	$Ra^{226} + 1/2$ dr.	4×10^{-6}
Cr^{51}	50	Rb^{86}	3×10^{-1}
$Cs^{137} + Ba^{137}$	1×10^{-1}	$Rn^{222} +$ dr.	2×10^{-4}
Cu^{64}	8	$Ru^{106} + Rh^{106}$	10
F^{18}	90	Sr^{89}	7×10^{-3}
Fe^{55}	4×10^{-1}	$Sr^{90} + Y^{90}$	8×10^{-5}
Fe^{59}	1×10^{-2}	Th-natural	5×10^{-5}
H^3(HTO or T_2O)	7	U-natural	7×10^{-3}
I^{131}	1×10^{-3}	Xe^{133}	4×10^{-1}
K^{42}	1	Xe-135	1×10^{-1}
Mn^{56}	15	Y91	20
Mo^{99}	1400	Zn^{65}	6

APPENDIX B

SUMMARY OF RADIONUCLIDE ANALYSES ON NAUTILUS AND SEAWOLF REACTOR COOLANT

PART 1 USS NAUTILUS (All Activities in μc/cc)

	15 min. Gross	120 Hr Gross	Mn^{56}	Co^{60}	Fe^{59}	Ni^{65}	Cr^{51}
Dumping Tolerance	3.0	---	15	2	1×10^{-2}	1.9	50
Date							
May 1955	1.4×10^{-1}	4.9×10^{-4}	9.3×10^{-2}	8.1×10^{-5}	2.6×10^{-5}	--	1.2×10^{-5}
Jun 1955	8.2×10^{-2}	8.9×10^{-3}	2.0×10^{-2}	2.7×10^{-3}	8.9×10^{-4}	1.8×10^{-4}	—
Jul 1955	8.2×10^{-2}	3.0×10^{-3}	1.3×10^{-2}	9.1×10^{-4}	3.6×10^{-4}	5.5×10^{-4}	1.0×10^{-4}
Aug 1955	2.2×10^{-1}	3.3×10^{-3}	1.0×10^{-2}	1.2×10^{-3}	5.3×10^{-4}	8.9×10^{-4}	2.8×10^{-5}
Oct 1955	1.5×10^{-1}	1.0×10^{-2}	7.3×10^{-3}	2.5×10^{-2}	2.8×10^{-3}	5.0×10^{-4}	3.4×10^{-3}
Nov 1955	4.5×10^{-2}	1.4×10^{-3}	9.8×10^{-3}	6.1×10^{-4}	1.1×10^{-4}	9.3×10^{-5}	1.6×10^{-5}
Dec 1955	1.3×10^{-1}	3.3×10^{-3}	3.4×10^{-2}	9.9×10^{-3}	1.5×10^{-3}	1.3×10^{-3}	6.5×10^{-4}
Feb 1956	8.1×10^{-2}	1.2×10^{-3}	9.2×10^{-3}	5.6×10^{-4}	7.4×10^{-5}	3.8×10^{-4}	2.4×10^{-5}
Apr 1956	1.3×10^{-1}	3.6×10^{-2}	6.4×10^{-2}	1.6×10^{-3}	2.2×10^{-4}	1.5×10^{-3}	9.1×10^{-5}
Jun 1956	7.0×10^{-2}	9.0×10^{-4}	1.3×10^{-2}	1.4×10^{-4}	4.1×10^{-5}	2.3×10^{-4}	5.9×10^{-6}
Jul 1956	4.7×10^{-2}	4.4×10^{-4}	4.5×10^{-3}	3.8×10^{-5}	2.0×10^{-5}	2.6×10^{-4}	5.9×10^{-6}
Aug 1956	1.5×10^{-1}	4.8×10^{-2}	7.0×10^{-3}	8.3×10^{-3}	1.7×10^{-3}	1.2×10^{-3}	1.3×10^{-3}
Oct 1956	4.6×10^{-2}	2.5×10^{-3}	1.1×10^{-2}	1.2×10^{-3}	1.6×10^{-4}	5.0×10^{-5}	1.4×10^{-5}
Dec 1956	5.1×10^{-2}	1.2×10^{-3}	1.8×10^{-3}	6.5×10^{-4}	9.0×10^{-5}	2.8×10^{-4}	8.8×10^{-5}
Feb 1957	5.8×10^{-2}	1.4×10^{-3}	—	5.8×10^{-4}	5.1×10^{-5}	--	3.0×10^{-5}
Mar 1958	2.3×10^{-2}	--	--	4.5×10^{-4}	4.8×10^{-5}	—	8.2×10^{-5}
Aug 1958	--	--	--	2.1×10^{-4}	1.7×10^{-5}	--	4.3×10^{-5}

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PART 1 Continued USS NAUTILUS (All activities in μc/cc)

	Na^{24}	Cu^{64}	Ta^{182}	F^{18}	W^{187}
Dumping Tolerance	3×10^{-1}	8	10	90	9×10^{-2}
Date					
May 1955	2.0×10^{-4}	2.9×10^{-4}	—	—	4.7×10^{-3}
Jun 1955	1.0×10^{-4}	3.4×10^{-4}	5.2×10^{-5}	5.5×10^{-3}	3.6×10^{-3}
Jul 1955	2.0×10^{-4}	6.8×10^{-4}	1.4×10^{-3}	2.5×10^{-2}	1.3×10^{-4}
Aug 1955	2.1×10^{-4}	1.8×10^{-3}	6.8×10^{-3}	—	1.1×10^{-3}
Oct 1955	5.0×10^{-4}	2.1×10^{-3}	2.9×10^{-2}	6.8×10^{-3}	1.3×10^{-3}
Nov 1955	5.4×10^{-5}	1.9×10^{-4}	1.2×10^{-3}	8.2×10^{-3}	1.4×10^{-5}
Dec 1955	6.2×10^{-4}	1.4×10^{-3}	1.7×10^{-2}	1.5×10^{-2}	9.5×10^{-4}
Feb 1956	2.5×10^{-4}	1.4×10^{-4}	1.8×10^{-4}	2.7×10^{-2}	1.5×10^{-4}
Apr 1956	4.1×10^{-4}	3.0×10^{-3}	4.5×10^{-3}	6.8×10^{-3}	7.7×10^{-4}
Jun 1956	4.8×10^{-5}	7.9×10^{-5}	5.7×10^{-5}	1.9×10^{-2}	1.8×10^{-5}
Jul 1956	4.9×10^{-5}	3.5×10^{-5}	7.6×10^{-5}	8.6×10^{-3}	1.8×10^{-4}
Aug 1956	2.7×10^{-4}	4.7×10^{-3}	2.2×10^{-3}	3.0×10^{-2}	9.0×10^{-3}
Oct 1956	1.1×10^{-4}	1.0×10^{-4}	1.8×10^{-5}	5.1×10^{-3}	1.3×10^{-3}
Dec 1956	1.0×10^{-4}	9.3×10^{-6}	6.7×10^{-6}	2.0×10^{-2}	3.0×10^{-4}
Feb 1957	—	1.1×10^{-4}	3.0×10^{-5}	—	—
Mar 1958	—	—	—	—	1.5×10^{-5}
Aug 1958	—	—	—	--	4.1×10^{-6}

Mar 1958	—	—	—	—	1.5×10^{-5}
Aug 1958	—	—	—	—	4.1×10^{-6}

PART 2 USS SKATE (All activities in μc/cc)

	15 min. Gross	120 Hr Gross	Cr^{51}	Fe^{59}	Co^{58}	Co^{60}	Mn^{54}	Zr-Hf	W^{187}
Dumping Tolerance	3.0	—	50	1.0×10^{-2}	2	2.0	1.3×10^{-1}	4×10^{-1}	9.0×10^{-2}
Date									
Oct 1957	1.7×10^{-2}	1.1×10^{-4}	2.5×10^{-6}	1.7×10^{-7}	1.4×10^{-7}	1.0×10^{-7}	1.6×10^{-6}	—	1.8×10^{-3}
Nov 1957	1.1×10^{-2}	7.5×10^{-4}	4.5×10^{-7}	2.4×10^{-6}	3.6×10^{-6}	2.2×10^{-6}	2.5×10^{-5}	1.2×10^{-7}	2.3×10^{-3}
Jan 1958	2.3×10^{-2}	1.7×10^{-4}	2.7×10^{-5}	2.4×10^{-5}	3.2×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	6.4×10^{-7}	3.1×10^{-3}
Feb 1958	1.7×10^{-2}	2.3×10^{-4}	7.3×10^{-7}	4.6×10^{-6}	7.0×10^{-6}	4.9×10^{-6}	3.7×10^{-5}	3.8×10^{-8}	2.1×10^{-3}
Mar 1958	3.2×10^{-2}	2.3×10^{-4}	1.5×10^{-5}	1.3×10^{-5}	2.4×10^{-5}	1.7×10^{-5}	5.1×10^{-5}	2.8×10^{-7}	2.9×10^{-3}
May 1958	8.0×10^{-3}	9.3×10^{-4}	3.0×10^{-5}	3.1×10^{-5}	7.6×10^{-5}	1.1×10^{-4}	6.5×10^{-5}	3.6×10^{-6}	2.1×10^{-3}

DEVELOPMENT OF A SELF-CONTAINED SCHEME FOR LOW-ACTIVITY WASTES.

by

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* E. Glueckauf

I. Statement of the Problem.

1. The disposal of liquid wastes which may be contaminated by radio-active materials is one of the major problems facing the atomic age.

2. While great strides have already been made in many directions, involving partial containment combined with partial dispersal, the principle of complete containment of radio-active waste has many advantages to offer in countries with a high population density.

3. It avoids the complaints, justified or not, that dissipated radio-activity could produce a health hazard to man or beast. It avoids situations whereby foodstuffs originating from certain areas are unjustly discriminated against due to public prejudice arising from ignorance of the technical facts. Moreover, it is a reassuring factor, if substances which are inherently toxic are kept under the permanent control of a body of persons, who are familiar with their characteristics, who do not object to their proximity, and who can judge the existence or non-existence of hazards, and act accordingly.

4. The article presented here will deal with the treatment of waste arising from research stations and power plants as opposed to the waste from fuel-processing plants. The latter usually deals with quantities of 10^7 curies per year, but their number is small, and their waste problems can be eased by carefully selecting the sites of location. The former may deal with activity levels which are more than 1000 times smaller, sometimes very much less than that, but the number of these sites in the world may become quite large in the future, and they may be located in or near highly populated areas.

5. The waste arising from such institutions consists mainly of three types.

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1) solid waste mostly from contaminated articles

2) liquid waste of small volume and comparatively high activity

3) liquid waste of very low activity and very large volume.

6. The problems in the containment of solid waste are trivial, but the practical application may be expensive on account of the large volumes accumulating over the years. Thus, it is often necessary to reduce this volume to the maximum extent, by incineration of all combustible material.

7. The problems presented by the containment of small amounts of liquid waste of moderately high activity are also negligible, and the volumes involved can be further reduced by evaporation to a saturated solution, or to dryness.

8. The main problem is thus the one where very large volumes of water are concerned which may contain little or even no activity, but which may be potentially contaminated above the permissible level. In every case of this kind, the problem is to decontaminate the water to such an extent that it can either be discharged into natural water channels or be re-used on the site, in which case it will also find its way eventually into the natural surroundings. The extracted activity can then be retained on the site, either in the form of concentrated solutions, or in the form of solids.

9. If one considers here the possibility of self-contained sites, such storage will be permanent. As it is apparent that solids can be stored with less risk of dissipation than liquids, the final aim must be, to reduce all waste into two final components:

1. Adequately pure water

2. Solid radio-active waste.

The most effective way to carry out this separation is by total evaporation of the waste water. But it is also the most expensive way. It can obviously be applied only where waste steam is plentiful in comparison to the quantity of liquid waste arising.

10. However, in large research stations the quantities of water involved may be of the order of 1000 tons per day, and with these quantities, chemical treatment schemes are quite generally more economical than evaporation.

11. The nature of the dilute aqueous wastes is of importance for the processes which can be chosen. Generally, the water will contain - in addition to soluble activity -

1) dissolved solids varying from 0 to 500 ppm.

2) Chemicals, such as acids, alkalis, organic matter including detergents.

3) Suspended solids and colloids.

12. The radioactive contamination may be present in the form of 1) cations, 2) anions, 3) non-ionic solutes, 4) colloids, and 5) suspended matter.

13. The choice of processes which will separate these constituents from the water, will depend to some extent on the purity to be achieved. The maximum permissible levels in water for occupational workers, recommended by the International Commission on Radiological Protection vary with each individual isotope. If the general public is likely to be affected the Commission advise a further safety factor of ten should be applied. The British Medical Research Council, on the other hand, advocate that, where large centres of population are concerned, a factor of one hundred should be used for all the more toxic isotopes. Probably the most restrictive levels laid down, so far, are those governing discharges to the river Thames. The maximum permissible levels agreed for this river water are:-

Radium	4×10^{-10} µc/ml.
other α emitters	2.4×10^{-9} µc/ml.
radio calcium and radio strontium	2×10^{-8} µc/ml.
other β emitters	1×10^{-6} µc/ml.

and these are considered as mutually exclusive.

14. These levels are, in many cases, lower than those of natural waters. For example the radium content of certain Scandinavian rivers can be as high as 6.5×10^{-7} µc/ml. Even certain large rivers such as the Nile (Africa) and the Plate (South America) contain up to ten times the radium level permitted in the Thames.

15. Apart from total evaporation of the water, there is no single process which will permit the conversion of a dirty radio-active effluent to drinking water. A multi-stage process is thus required.

16. The presence of suspended solids with its adsorbed radio-active matter makes a filtration step essential. It is therefore convenient to increase the precipitable matter, by addition of chemicals which will result in a well-settling floc. This stage alone can result in a clear fluid, with a decontamination which permits discharge into the Thames. After a single-stage precipitation treatment, most of the caesium and iodine, some strontium, and some ruthenium, usually survive in the fluid. Further stages are thus required. These can have the form of further precipitations, preferably combined with treatments intended to convert anionic and non-ionic ruthenium species into cationic precipitable ones. Other types of second-stage treatment which can be carried out cheaply, are a) passage through an ion exchange bed of a type which is highly specific to strontium and caesium removal, or b) complete deionisation. The means to do this are discussed in the subsequent paragraphs.

IIa. Chemistry of the Precipitation Processes.

17. Much work has been carried out at Harwell during the past ten years both

in the laboratory and on the plant to establish the most satisfactory flocculation treatment. Table 1 gives details of some of the results obtained on the original plant which was designed in 1946. The activity removals given are the averages of many hundreds of results.

Table 1.

Treatment	pH	Unspecified α removal	Unspecified β removal
Tannic acid + Lime + Trisodium phosphate (40 ppm as PO_4)	9.4	98.5%	75%
Trisodium phosphate (100 ppm) + Lime or Sodium Hydroxide	9.5	95.0%	75%
Alum (20 ppm Al) + Lime or Sodium Carbonate	8.0	89.0%	70%
Alum (40-80 ppm Al) + Activated Silica (5-10 ppm SiO_2)	8.0	80.0%	65%

18. It has been proved at A.E.R.E. and by many other workers in this field that the decontamination factor for both α and β emitters is largely dependent on the pH value used. In general the higher the pH value the greater the efficiency of removal with an optimum at about 11.5 (see Fig. 1). However if the effluent is discharged finally to rivers or sewers or the sea a restriction in final pH value is often enforced, a value between 7.0 and 8.5 being required. A high initial value therefore gives rise to higher treatment costs in that partial neutralisation is required and, in addition the total dissolved solids content may be raised, thereby, to an undesirable high level. It is for these reasons that the pH value has been restricted in the past to approximately 9.5 or below. Although only 75%, on the average, of the unspecified β activity was removed by working at this figure the decontamination was sufficient to enable the large volume low level liquid wastes to be discharged within the authorisations granted.

19. As the early work had established that the phosphate treatment was the most suitable it was decided to concentrate on this process for the first step in the development work directed towards a self contained scheme. The reaction is primarily between the calcium present in, or added to the effluent with the ortho-phosphate added. As mentioned earlier the efficiency of removal of the radioactive isotopes increases with the pH value used up to approximately 11.5 (see Fig. 1). All recent work described below has, therefore, been carried out at this value.

20. The effluent arising at Harwell contains appreciable amounts (ca 100 ppm.) of natural calcium due to the hardness of the water used in the Establishment. If this is not the case, as in certain special effluents, a minimum of 50 ppm. of calcium must be added. It is most convenient to add this as a soluble calcium salt. Preliminary ideas were to raise the pH value with calcium hydroxide but apart from the difficulty of attaining the required value the use of lime gives rise to an increased amount of sludge and one that is difficult to control. In consequence it has been found advisable to use sodium hydroxide.

21. Concerning the composition of the calcium phosphate formed, available information suggests that a hydroxy-apatite is formed of probable composition $3\ Ca_3(PO_4)_2\ Ca(OH)_2$. The ortho-phosphate used is commercial tri-sodium phosphate ($Na_3PO_4 . 12\ H_2O$) and numerous experiments have shown that a minimum dose of 80 ppm. as PO_4 is required. With the techniques used little advantage is to be gained by increasing the phosphate concentration much above this value; in fact excess will give rise to profuse algal growth which not only tends to concentrate the radioactive isotopes but will also be a serious embarrassment in subsequent stages of the treatment.

22. Although the phosphate floc produced under the above conditions is in many cases satisfactory it has been found that there are certain interfering constituents which may be present in the effluent that tend to prevent the formation of a heavy sludge. Small amounts of detergents, for example, have been shown to prevent settling. It has been proved that a denser precipitate is formed if a ferric salt is added as a conditioning agent. It is generally accepted that under such conditions ferric oxide should be formed together with ferric phosphate as $x\ Fe_2O_3\ y FePO_4 . 3\ H_2O$. With the dosages used in this work the values of x and y are probably 3 and 1 respectively.

23. Experimentally it has been found that a ferric salt is preferable to a ferrous, giving a more controllable precipitate and therefore better removals. This is unfortunate owing to the corrosive nature of ferric salts and their comparatively high costs. Good results have been obtained with ferrous sulphate provided there is sufficient oxygen in the effluent to complete oxidation or by deliberately oxidising the ferrous salt just prior to addition by chlorine. One further advantage of the addition of an iron salt is that any excess phosphate is removed. The most satisfactory dosage of iron salts has been shown to be 40 ppm. as Fe^{3+} in conjunction with the other chemicals as quoted.

24. The combined calcium-iron-phosphate process in the sludge blanket filters has given consistently good results on the effluent treated. The average removal over twenty five consecutive pilot plant runs is shown in Table 2.

Table 2.

	Phosphate treatment % removal	Phosphate treatment followed by sulphide treatment. % removal
α emitters	98.78	99.65
β emitters	86.91	90.89
radio ruthenium	47-70	69-83
radio caesium	16-33	32-49

25. Although the results of the phosphate treatment are satisfactory it was decided to try out a second precipitation stage with the hope of increasing the decontamination obtained. With ruthenium particularly in mind a sulphide precipitation appeared attractive. This process, which is carried out on the effluent from the phosphate treatment, consists of the addition of 20 ppm. of Fe^{2+} as ferrous sulphate and 20 ppm. of S^{2-} as sodium sulphide. As will be seen from Table 2 the activity removal is increased by this stage. The efficiency of the treatment for radio ruthenium is variable depending on the form present but the process does appear effective in removing radio colloids and in entrapping light floc particles carried over from the phosphate treatment.

26. Contrary to expectations it has been found that between 16 and 33% of caesium is removed by the phosphate precipitation, rising to 32-49% after sulphide treatment (see Table 2).

IIb. Practical Aspects of the Precipitation Stage.

a) Sludge blanket clarifiers.

27. The main effluent treatment plant at Harwell was designed in 1946 and has been described in detail elsewhere(1 & 2). In brief it consists of pre-treatment batching tanks, chemical dosing tanks, clarifiers, and post-treatment tanks.

28. The success of any flocculation treatment depends very largely on the performance of the precipitator used. The clarifiers on the existing plant rely,

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(1) Wilson, C. L., "Design and Construction of a Handling and Treatment System for Liquid Radioactive Wastes", Proc. Inst. Civil Eng., 4, 1, 1 (1955).

(2) Burns, R. H., "Operational Experience with a Handling and Treatment System for Liquid Radioactive Wastes", Proc. Inst. Civil Eng., 4, 1, 21 (1955).

almost entirely, on settling of the sludge formed. Recent work in this field indicated that a sludge blanket type of precipitator could be expected to give better results. In consequence small filters of this type were manufactured for laboratory experiments, and large plant purchased for the pilot plant work. In these (see Fig. 2) the effluent is pumped to a flash mixing chamber where the liquid and chemical reagents are mixed by a high speed motor driven agitator. After treatment the effluent passes downwards into the flocculation zone of the precipitator where a slow agitator ensures complete circulation and suspension of the chemical precipitate thus promoting a growth of large floc particles. From the flocculation zone the liquid passes without violent change in direction under an inclined baffle wall into the filtration zone where the upward velocity is progressively reduced resulting in the formation of a sludge filter blanket by the intimate contact of fine precipitate particles with previously formed sludge. This arrangement facilities the growth of the particles and increases their density. A stable blanket is formed thereby which can resist the upward movement of the liquid. Any heavier particles will subside to the bottom of the precipitator.

29. The thickened sludge is removed at intervals in sharp bursts of adjustable length to suit operating conditions, thus maintaining a constant depth and density of the sludge blanket.

30. The clarified effluent issuing from the sludge filter blanket rises to the top of the precipitator where it is collected in a sparge pipe and led to a settling chamber.

31. In general very satisfactory results have been obtained using apparatus of this design; a clear supernatant being obtained. One disadvantage is the presence of mechanical parts, needing maintenance, within the active zone. In an endeavour to obviate this experiments are being carried out at the present time with a "Pulsator" unit (see Fig. 3). The principle of a sludge blanket filter is still used but the mixing of the effluent and reagents, the ageing of the floc so formed and the control of the blanket is carried out by introducing the treated effluent in intermittent bursts rather than as a continuous flow. Initial tests indicate that a clear supernatant is obtained, and that the throughput is approximately 25% greater than for the more conventional type of plant.

β) Treatment of Chemical Sludges

32. The chemical sludges removed from the precipitators may average 3% suspended solids. It is necessary to concentrate these in order to effect economy in disposal or storage. The phosphate - sulphide sludges produced are difficult to filter, do not settle well and the final sludge cakes produced usually contain up to 84% water. Centrifuging gives no better results and, in addition, the centrifuged effluent is often unsatisfactory as regards clarity. It was for this reason that a freezing treatment has been adopted at Harwell. In this process the settled sludge is frozen then thawed and the resultant liquor filtered on a vacuum filter. A flow diagram of the plant is shown in Fig. 4.

33. The main features and requirements of the process are outlined below:-

(1) The sludge must be completely frozen. Even a small percentage of material which has been left unfrozen will impair seriously the efficiency of the process. The plant at Harwell operates at -2°C.

(2) The sludge must be frozen slowly. This gives rise to large crystals of ice ensuring the maximum benefit from the treatment. Flash freezing must be avoided.

(3) The rate of thawing is not important.

(4) Filtering of the thawed material should take place as soon as possible after the thawing cycle is complete (i.e. within 8 hours).

(5) Vigorous agitation of the thawed sludge should be avoided.

(6) Better results are obtained by freezing, thawing and settling than by conventional filtration methods applied to the untreated sludge.

(7) The process does not release activity from the treated sludge.

34. The mechanism of the process appears to be that the freezing concentrates the electrolytes present around the colloidal sludge particles giving rise to coagulation. The final solid matter is granular in character and settles and filters well. The solids content of the sludge cake after filtering is often twice that obtained by straight filtration of the untreated sludge and the final sludge volume is approximately 50% of that normally obtained.

35. Table 3 gives some typical results obtained with various types of filters during the preliminary development work.

36. The filtration arrangements on the existing large scale plant consist of a nylon belt filter to which vacuum may be applied. The solids content of the final sludge is approximately 47% higher than the corresponding value for the filter cake obtained from the pressure filter which is used as a standby and deals with untreated sludge. The sludge volume is approximately 57% lower.

37. The improvement in filtering rate caused by the freezing and thawing treatment is most remarkable (see Fig. 5).

Table 3.

Type of filter	Volume filtered	Volume of Sludge Cake		Time Taken		Solids Content of Final Cake	
		U* ml.	T† ml.	U* hours	T† mins.	U* %	T† %
Small laboratory pressure filter 5 lbs./sq.in.	250 ml.	80	40	0.6	2	11.5	40
Small laboratory pressure filter 5 lbs./sq.in.	250 ml.	50	30	7	2	8.8	39
Pressure filter 30 lbs./sq.in.	182 litres	0	13,600	-	5	-	32
Combined Pressure and Vacuum	6 litres	3,500	2,300	24	15	15.6	49
Settlement alone Original solids 6.5%	250 ml.	240	65	48	1	6.8	21.5

* Untreated

† Treated

III. Inorganic Ion Exchangers.

38. After the flocculation treatment, the only active fission product cations remaining in solution are Cs^{+} and a small quantity of Sr^{++}. To remove these quantitatively from a solution of pH 11 which is about N/100 with respect to sodium ions, various ion exchange methods were investigated. The use of synthetic ion exchangers would only be economical, if it is possible to regenerate the exchanger bed. But regeneration would create again active solution, which moreover, would contain a very large amount of solid regenerants (NaOH and H_2SO_4) and which would represent another inconvenient disposal problem.

39. As an alternative, inexpensive naturally occurring inorganic ion exchangers have been considered. These are obtainable at low prices and can be treated as expendable, without the need for regeneration. They would still represent a storage or disposal problem, to which, however, experimental approaches offer a promising solution.

40. Both clays and vermiculite have been investigated. While the former, notably montmorillonite, has a relatively high exchange capacity (1.0 eq.kg.) its mechanical filtration properties are such that it is only useful after some kind of processing which produces a granular structure(3,4 & 5).

41. By comparison, vermiculite, both in its natural and expanded form, has excellent filtration properties, permitting linear filtration velocities in excess of 1 m. per hour. Vermiculite is a magnesium or calcium-magnesium alteration product of mica with the composition

$$(H_2O)_x \cdot (MgCa)_y \cdot (Al\,Fe\,Mg)_z \cdot (Si\,Al\,Fe)_4O_{10}\,(OH)_2 \cdot$$

The exact values of x, y and z appear to be variable. Cation exchange capacity figures are quoted as high as 1.34 - 1.46 eq./kg. but for the crude material used at A.E.R.E. the figure lies between 0.55 - 0.75 eq./kg. Vermiculite has a high specificity for the exchange of both Cs and Sr ions, in the presence of sodium salts.

42. Table 4 shows the breakthrough volumes for caesium and strontium for vermiculite columns under varied experimental conditions in the presence of N/100 Na solution.

Table 4.

Ionic solution percolated	Breakthrough volume (litres) per Kg. of vermiculite	Breakthrough volume (litres) per Kg. of organic exchanger
Cs tracer	3,000 ----	{ 600 (mixed bed) { 1,500 (Zeo-carb 225)
Cs, 0.16 ppm.	400	–
Cs, 0.16 ppm. ($+NH_4$ 5 ppm.)	400	–
Sr, 1.5 ppm.	800	–

(3) Ginell, W. S., Martin, J. J. and Hatch, L. P., "Ultimate Disposal of Radioactive Wastes", Nucleonics, 12, 12, 14 (1954).

(4) Hatch, L. P. and Regan, W. H., "Concentrating Fission Products", Nucleonics, 13, 12, 27 (1955).

(5) Beirne, T., Grover, J. R. and Hutcheon, J. M., "Bonded Montmorillonites for the Ultimate Disposal of Long-lived Fission Products", Harwell Report, A.E.R.E. CE/R-1658 (1955).

43. Thus vermiculite compares well with the organic exchangers, a fact which is due to the higher selectivity of the vermiculite for strontium and caesium.

44. Harwell experiments have shown that up to the breakthrough point, a decontamination factor of 5 x 10^3 to 10^4 was obtainable with a bed depth of 15 cm. and a linear flow rate of 80 cm./hour, with exfoliated vermiculite. Markedly better factors are obtainable with deeper beds.

45. Four vermiculite columns have been installed in the Harwell plant, two designed for downward flow, and two for upward flow. The vermiculite beds are 19.0 sq. feet area and 16 inches deep and are supported on an eight inch bed of graded gravel. The volume of the beds is approximately 158 gallons and the weight of vermiculite used is about 1 ton per column. The optimum flow is 16 gph per sq. foot when dealing with an effluent containing 350-400 ppm sodium. This allows a 40 minute contact time within the column. A higher flow rate, even if attainable is undesirable as experiments have shown that the exchange rate is slow.

46. The effluent from the plant columns did not show an increase in activity level, till approximately bed volumes had been passed. This figure is characteristic of the sodium ion concentration in the feed (350-400 ppm.) and of the content of 16 ppm. calcium ion, which remains in the effluent from the sulphide precipitation stage. Larger breakthrough volumes can be expected for solutions containing less electrolyte.

47. Table 5 gives the results for the last 25 runs on the plant.

Table 5

	Removal of α emitters (% age)		Removal of β emitters (% age)	
Phosphate followed by Sulphide Treatment	99.65	max. 99.92 min. 99.47	90.89	max. 94.54 min. 86.58
Followed by vermiculite columns	99.95	max. 100 min. 99.83	99.36	max. 99.67 min. 99.17

48. The residual β-activity (0.64%) is mostly due to anionic and non-ionic compounds about half being due to ruthenium.

49. The activity level of the large volume effluent arising at Harwell varies considerably but on the average amounts to 2.7 x 10^{-5} μc/cc. of α emitter and 1.8 x 10^{-5} μc/cc. of β emitters. This low figure is only maintained by segregating from the main effluent stream those liquids known to be appreciably active. Reference to Table 5 shows that approximately 99.95% of the α emitters and 99.36% of the β emitters can be removed by the precipitation and column treatment. The resulting levels viz. 1.4 x 10^{-8} μc/cc. for α activity and

1.2×10^{-7} μc/cc. for β emitters are obviously safe and mean that effluent decontaminated to this level could be re-used for certain purposes within the Establishment. The high salt content would preclude its use as drinking water.

50. The vermiculite, after draining, can be easily stored, with or without further treatment.

IV. Electrolytic Deionisation.

51. The process of electrolytic deionisation is analogous to that used for the de-salting of sea-water, where an electric current transfers the ionic constituents of water through two semi-permeable membranes, each passing ions of positive or negative charge only. The normal de-salting cells are useful for reducing the salt content of water to about 200 ppm. After that the resistance of the water increases too much, so that very high voltages would be required. This difficulty can be overcome by filling the central compartment with mixed bed ion exchanger beads, the conductance of which is much higher than that of dilute solutions(6).

52. The performance of this type of equipment can now be compared with that of a mixed bed ion-exchange column, where the column is continuously regenerated, by the electric current passing at right angles to the solution flow.

53. As the mixed bed contains both acid and alkaline exchanger particles, it cannot be used for the continuous removal of ions which form insoluble hydroxides, though very small quantities of calcium and strontium do not affect the performance. But it has been shown at Harwell that radio-activity contaminated waste solution, after an alkaline phosphate sludge treatment, can be deionised completely for an apparently indefinitely long time. Some anxiety was felt about the effect of detergents on the performance, but so far the tests have not shown any deleterious effects on the cells used or on current consumption.

54. In the case of the usual organic ion exchanger beds the activity remains with the highly concentrated regeneration fluids; in the case of non-regenerated inorganic exchanger it remains with the bulk of the material used. The advantage of electro-deionisation lies in the fact that no such dilution occurs, and the activity thus arises with the minimum of bulk.

55. The deionisation characterised by the effluent concentration C_e which can be achieved in a single cell unit, depends apart from the cell dimensions on the following operational parameters:

1) the concentration of the inflowing water, C_i (moles/litre).

2) The flow rate of the main stream F_i (litres/min.).

3) The current passing through the cell A (Amps.) and

4) The concentration C_w of the concentrated waste stream leaving the cell.

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(6) Walters, W. R., Weiser, D. W., Marek, L. J., "Concentration of radioactive aqueous wastes", Ind. & Eng. Chem., 47, 61 (1955).

56. Experimentally it was found that, if $C_e \ll C_i$, a simple relation holds between these parameters.

$$\frac{C_e A}{C_i C_w F} = \text{constant} = 0.6.$$

This relation holds for the particular cell in the region investigated which ranged over the values

$$4 \times 10^{-6} < C_e < 1.10^{-3}$$
$$7 \times 10^{-5} < C_i < 5.10^{-3}$$
$$1 \times 10^{-2} < C_w < 4.10^{-1}$$
$$2.5 < F_i < 3.5$$
$$0.5 < A < 3.1$$

57. The equation shows that the effluent concentration increases proportional to the concentration in the waste compartment and, as it is desired to obtain a minimum C_e and a maximum C_w without an unduly large current consumption A, it is necessary to run at least two cells in series, the waste compartments of the second one being fed with pure water, and discharging into the waste compartments of the first cell, see Fig. 6. Some performance data of this type of multiple cell are shown in Table 6.

Table 6.

Typical performance figures of individual cells (used in series).

Cell	C_i	C_e	C_w	Watts	KWh/equiv. required	$\frac{C_e \cdot A}{C_i \cdot F_i \cdot C_w}$
	moles per litre					
1	3.7×10^{-3}	8.2×10^{-4}	0.36	310	0.57	0.64
2	8.2×10^{-4}	1.3×10^{-4}	0.083	100	0.79	0.64
3	1.3×10^{-4}	5×10^{-6}	0.016	55	2.6	0.50
Overall:				465	0.71	

Flow rate of main stream F_i = 3 litres/min.

Flow rate of waste stream F_w = 0.03 litres/min.

A = Amperes (Watts used/100 V).

58. The efficiency of de-ionisation, η, can be measured by the power divided by the removed electrolyte. η was found to be proportional to the current density and inversely proportional to the waste concentration C_w. The power efficiency can therefore be increased, by using cells with a larger membrane area, but this increases the investment cost.

59. The cell system of Table 6 will deionise 1000 l./hour of water of 100 ppm. to a value of 0.1 ppm. with a power consumption less than 3 KW.

60. There is no difficulty in deionising the water completely, (with reduced current efficiency). Even with a single cell conductivity water of $3 \times 10^6\ \Omega$ has been obtained, which is equivalent to the product of small mixed bed deionisation columns.

61. It would appear that there is a considerable future for large scale deionisation cells for the final purification of the effluent after prior flocculation treatment. Pilot plant tests are at present in progress.

V. Fixation of activity in solid form.

62. The previously discussed treatments are all concerned with the purification of the large masses of water with a view to returning these, purified, to useful purposes or to their natural surroundings.

63. The activity which has been extracted and concentrated, still has to be dealt with. Having made a considerable effort to concentrate it, one would obviously wish to keep it so, and ensure that it cannot be dissipated again. The only way to ensure this is to convert the accumulated waste to a ceramic or glass, though, in arid areas, deep burial would appear to be an adequate form of disposal.

64. Several methods have previously been discussed for the conversion of active solutions into ceramics. There is the ion exchange adsorption on montmorillonite followed by heating at 900° (see Ref. 3 and 5), and the direct conversion of aluminum containing waste to a calcined material[4]. Other methods are the mixing of liquid waste with nepheline-ayenite followed by calcination at 1100°C[7], the mixing of waste with Conasauga shale followed by fusion or partial fusion at 1100°C[8], the drying of liquid wastes in fluidised beds[9], and finally the

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(3) See Footnote page 10.

(4) See Footnote page 10.

(5) See Footnote page 10.

(7) White, J. M. and Lahaie, G., "Ultimate Fission Product Disposal", Chalk River Report CRCE-591 (1955).

(8) Struxness, E. G., Morton, R. J. and Straub, C. P., "Disposal of High-level Radioactive Liquid Wastes in Terrestrial Pits", Proc. Int. Conf. on the Peaceful Uses of Atomic Energy. 9, p.684 (1955).

(9) Jonke, A. A., "A fluidized bed technique for treatment of aqueous nuclear wastes by calcination to oxides", USAEC Document TID-7517 (Part 1a) p.374 (1956).

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Harwell experiments covering a wide variety of fusion and leaching experiments both on a laboratory scale[10] and, on a pilot plant scale[11]. Fig. 7 shows some of the leaching tests by Grover and Furbey on fission product saturated clay, fired at 1250°C, which show that the product is resistant not only to water but also to acid solutions.

65. Most of this work was aimed at the conversion of high-activity waste, and a similar though much simpler problem arises also in the case of the low activity waste from water treatment. One would naturally aim at a single process which could deal with all the low-activity arisings from a research or processing laboratory. These consist of

1) dried sludges from precipitation treatments, mainly calcium and iron phosphates.

2) Ashes from the incinerator, largely calcium oxides or carbonates.

3) Evaporated concentrates of solutions collected from laboratories and of the effluent from the electro-deioniser, largely sodium nitrates, sulphates and other salts.

4) The vermiculite from exhausted filter beds which by this time will consist largely of magnesium-alkali-silicate.

It is apparent from the composition of these wastes that the addition of silica-sand and sodium carbonate or borate in the appropriate quantities will produce a mixture which can readily be transformed into a low-melting glass with adequate non-leaching properties.

66. In this field, a considerable amount of engineering development is still required, to effect the mixing of the various streams of solid and liquid waste to a homogeneous mixture, as well as the calcination and the final fusion to a glass.

67. Both calcination and fusion treatments must incorporate devices for trapping ruthenium from the gases and vapour emitted during the treatment. Amphlett's work has shown that the evaporated ruthenium fraction is only moderate, so that by means of recycling it, all the ruthenium can be incorporated in the glass.

68. Table 7 shows some of the results obtained by Amphlett and Warren[10].

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(10) Amphlett, C. B. and Warren, D. T., "Fixation of Activity in Solid Form by Absorption on Soils", Harwell Report AERE C/R-1686 (1956).

(11) Grover, J. R. and Furbey, P. R., "Ultimate Disposal of Long-lived Fission Products", Harwell Report AERE CE/R-2303 (1957).

Table 7.

Mix	Loading	Activity lost % of total	Approx. % of Ru
3N HNO_3 solution of F.P.'s (200 fold) + soil + 33% Na_2CO_3 + graphite	2 meq/g.	1.2	~25
Same solution precipitated with NH_3 + soil + 33% Na_2CO_3	2	1.2	~25
Same solution precipitated with NH_3 + soil + 33% $NaNO_3$	2	0.8	~16
Same solution + $NaNO_3$ (no soil)	~25	2.6	~50
Fired in Al_2O_3 crucible for 16 hours at 1000°C in N_2 atmosphere. Soil: Lower Greensand. Product: glass of density 2.4.			

VI. Layout of Integrated Treatment Schemes.

69. The present Harwell plant incorporating the unit processes outlined above is illustrated in Figure 8. The throughput is between 500 and 750 gallons per hour and the plant is dealing adequately with the daily effluent from certain of the radio chemical laboratories and process buildings. No great operational difficulties have been experienced and the system can be recommended where a good decontamination is required on low level liquid wastes. The preliminary laboratory and pilot plant work has been described elsewhere(12 & 13).

70. Three possible alternative schemes are considered in detail in Figure 9, 10 and 11, and summarised in Table 8. An attempt has been made to give costs for the various processes. These should be taken as comparative only as the actual values will vary widely from place to place.

71. It has been assumed that the total effluent to be treated is 100,000 gallons per day in all cases. Depreciation of plant and buildings has been taken as ten per cent of the capital cost and maintenance at three per cent except for certain specialised equipment when a figure of eight per cent has

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(12) Seedhouse, K. G., Monahan, J. and Wallis, G., "The removal of Fission Products from Solution with a Precipitator-Column Treatment. Part I. Laboratory Trials", Harwell Report AERE ES/R-2220 (1958).

(13) Seedhouse, K. G., Monahan, J., Pearce, K. W. and Wallis, G., "The removal of Fission Products from Solution with a Precipitator-Column Treatment. Part II. Pilot Plant Trials", Harwell Report AERE ES/R-2089, (1957).

been assumed. Loss of interest on capital has been calculated at three per cent and industrial labour charged at five shillings per hour. Additional costs have been added to cater for supervision and scientific control where appropriate.

72. The freezing treatment adds considerably to the cost of dealing with the chemical sludges produced. The value of £7.5 given in Figure 9, 10, and 11 for filtering 1,000 gallons of sludge would increase to approximately £10. The process is only of advantage therefore, if, as at Harwell, storage or ultimate disposal is difficult or very expensive.

73. The cost of the electro-deionisation plant has been assessed at £10 per gallon per hour throughput, and this was based on the cost of one experimental unit. It may well be that mass production would reduce substantially this figure and give a saving of approximately £8 per 100,000 gallons of effluent treated.

Table 8.

Scheme	Process	Decontamination obtained		Volume for Storage	Cost of 100,000 galls. of Effluent	Remarks
		α Emitters	β Emitters			
1	Precipitation and controlled disposal.	99.65%	90.89%	560 gallons	£123.25	The residual β activity is largely due to caesium with some strontium, ruthenium and iodine.
2	Precipitation and Vermiculite Columns	99.95%	99.36%	718 gallons	£117	The residual β activity is largely due to non-ionic ruthenium and anionic species such as radio-iodine.
3	Precipitation and Electro-deionisation.	100%	99.85% to 100%	685 gallons	£144.1	The residual β activity depends entirely on the amount of non-ionic ruthenium compounds.

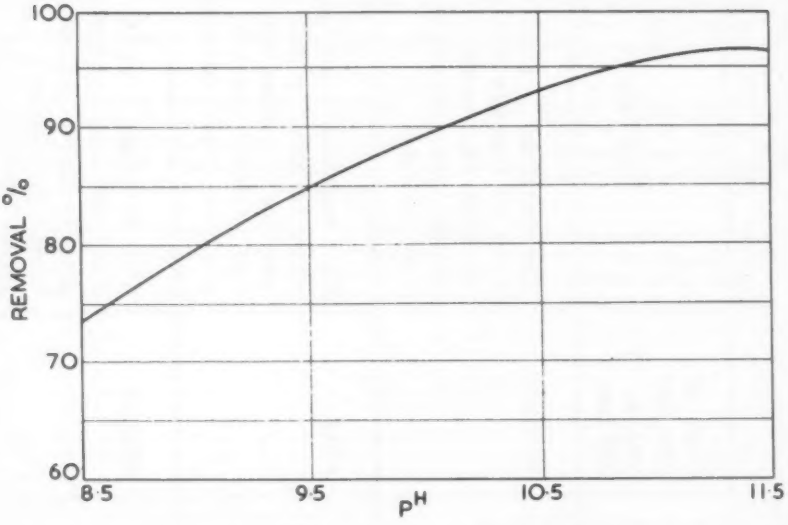


FIG. 1. REMOVAL OF MIXED FISSION PRODUCTS BY PHOSPHATE COAGULATION USING 80p.p.m. PO_4 20 Fe^{3+}

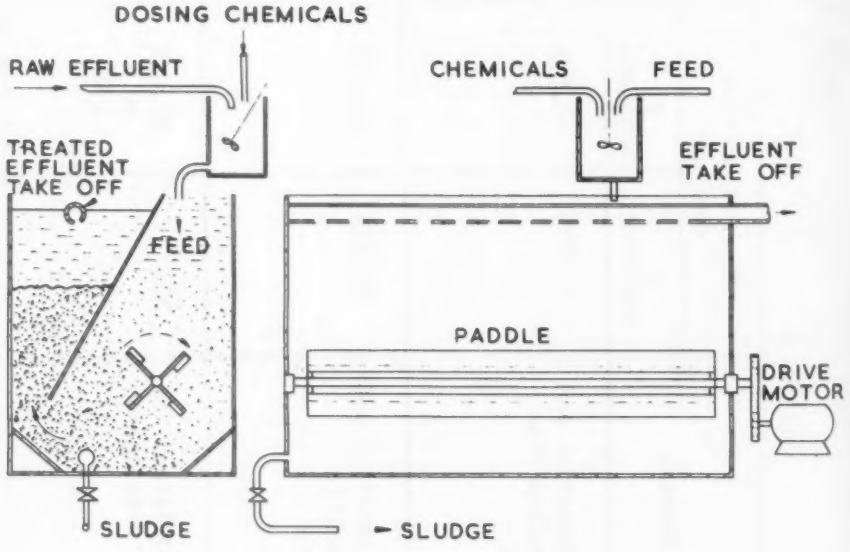
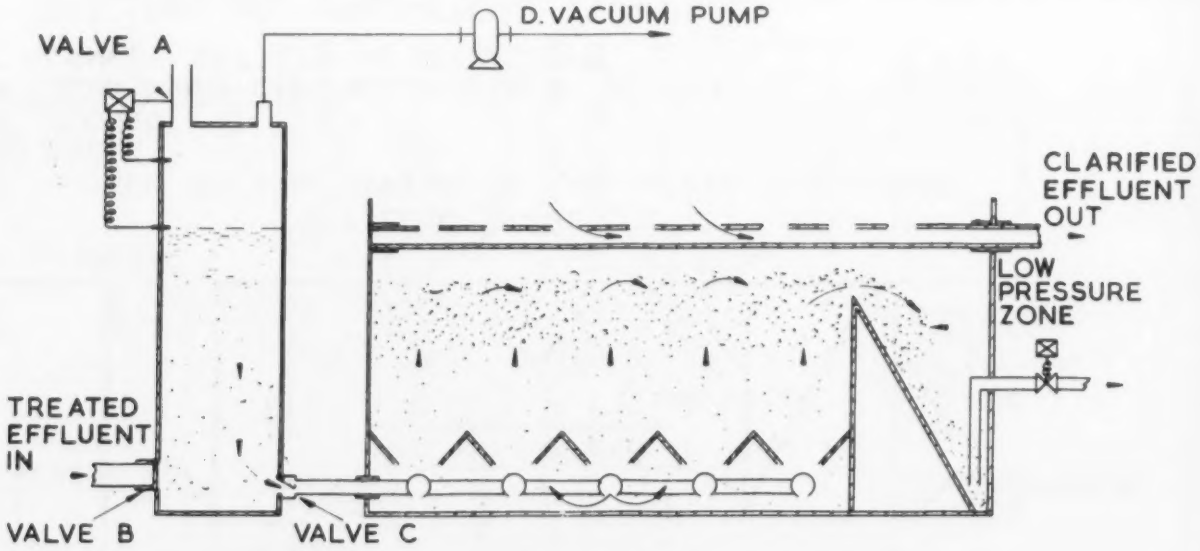


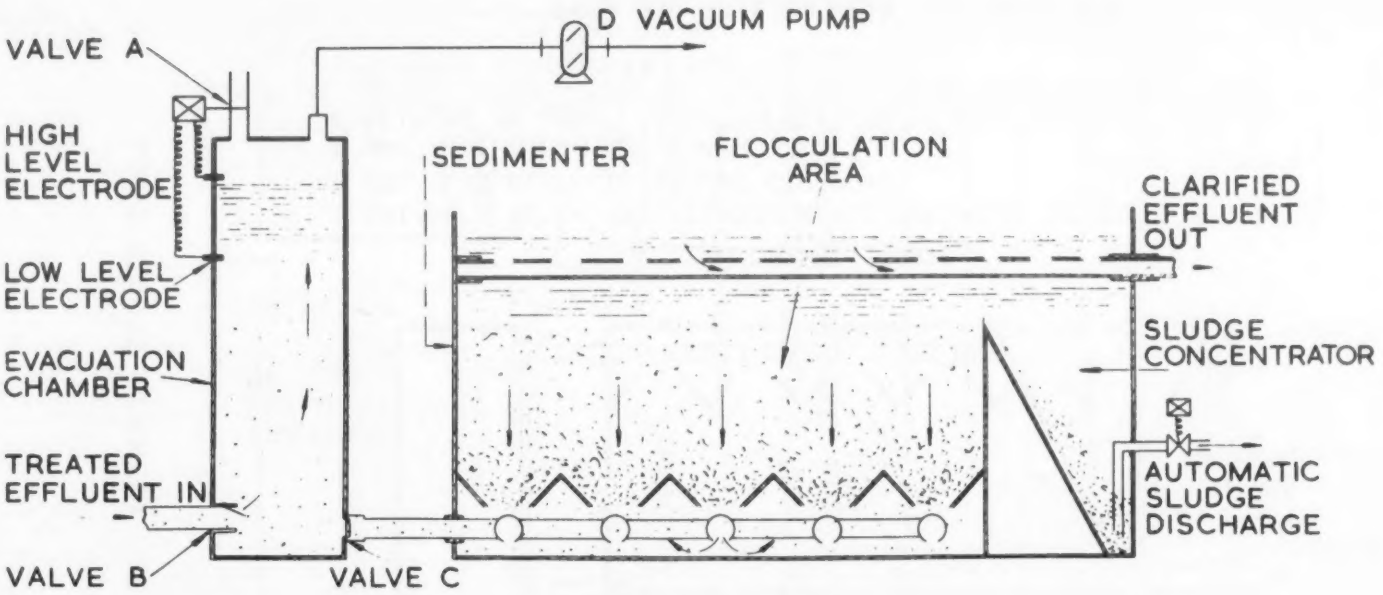
FIG. 2. SLUDGE BLANKET PRECIPITATOR.



PULSING STAGE

1. VALVE 'A' OPEN TO ATMOSPHERE BY HIGH LEVEL ELECTRODE.
2. VALVE 'B' CLOSED. VALVE 'C' OPEN.
3. VACUUM CHAMBER EMPTYING.
4. SLUDGE BLANKET COMPRESSED WITH THE EXCESS SLUDGE BEING ATTRACTED TO THE LOW PRESSURE ZONE IN THE SLUDGE CONCENTRATOR.

FIG. 3A. "PULSATOR" CLARIFIER.



SETTLING STAGE

1. VALVE 'A' CLOSED TO ATMOSPHERE BY LOW LEVEL ELECTRODE.
2. VALVE 'B' OPEN.
3. VALVE 'C' CLOSED.
4. VACUUM FROM PUMP CAUSING CHAMBER TO FILL.
5. SLUDGE BLANKET SETTLES IN SEDIMENTER.

FIG. 3B. "PULSATOR" CLARIFIER

FIG. 3B. PULSATOR CLARIFIER

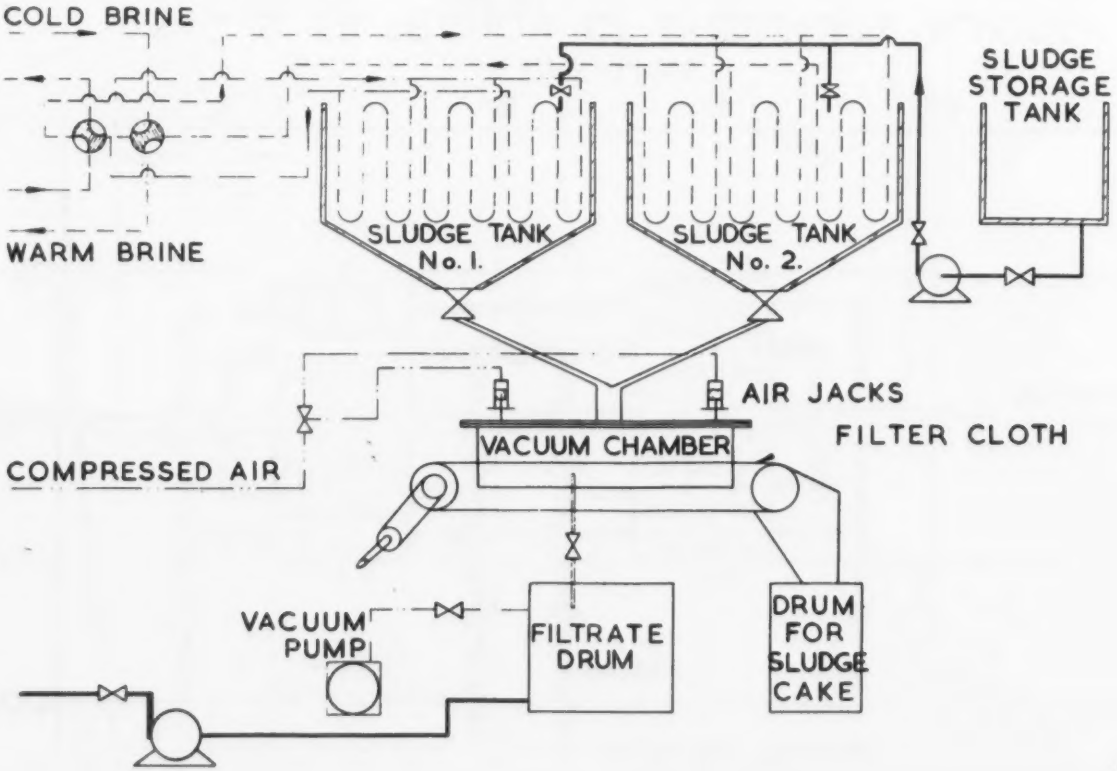


FIG. 4. FLOW DIAGRAM OF SLUDGE FREEZING.

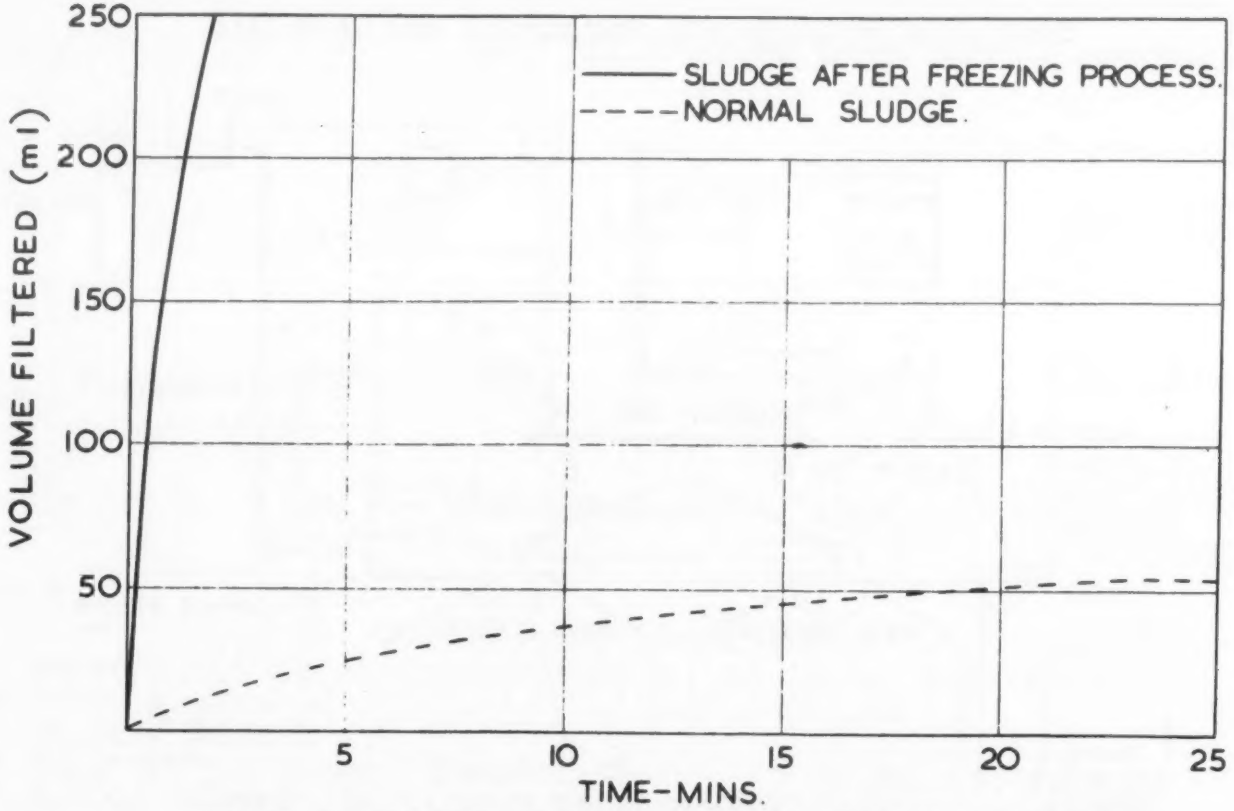
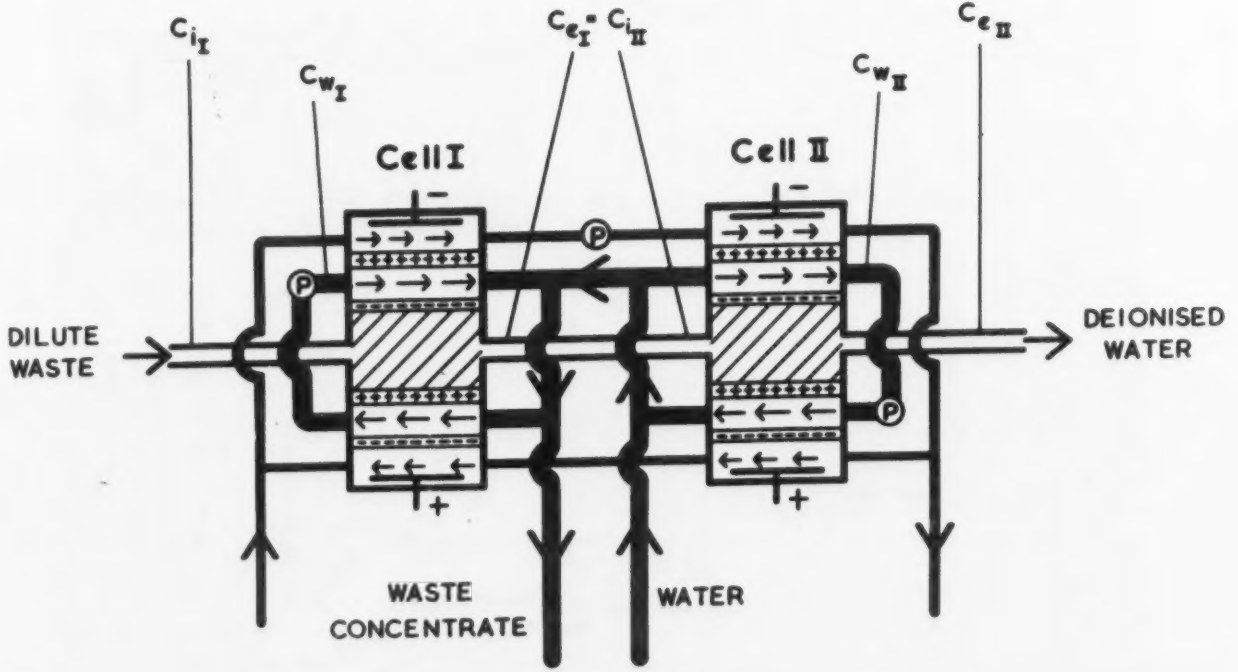


FIG. 5. FILTRATION OF PHOSPHATE SLUDGE.

FIG. 6.

FIG. 6.



FLOW DIAGRAM FOR A TWO-CELL ELECTRODEIONISATION TREATMENT.

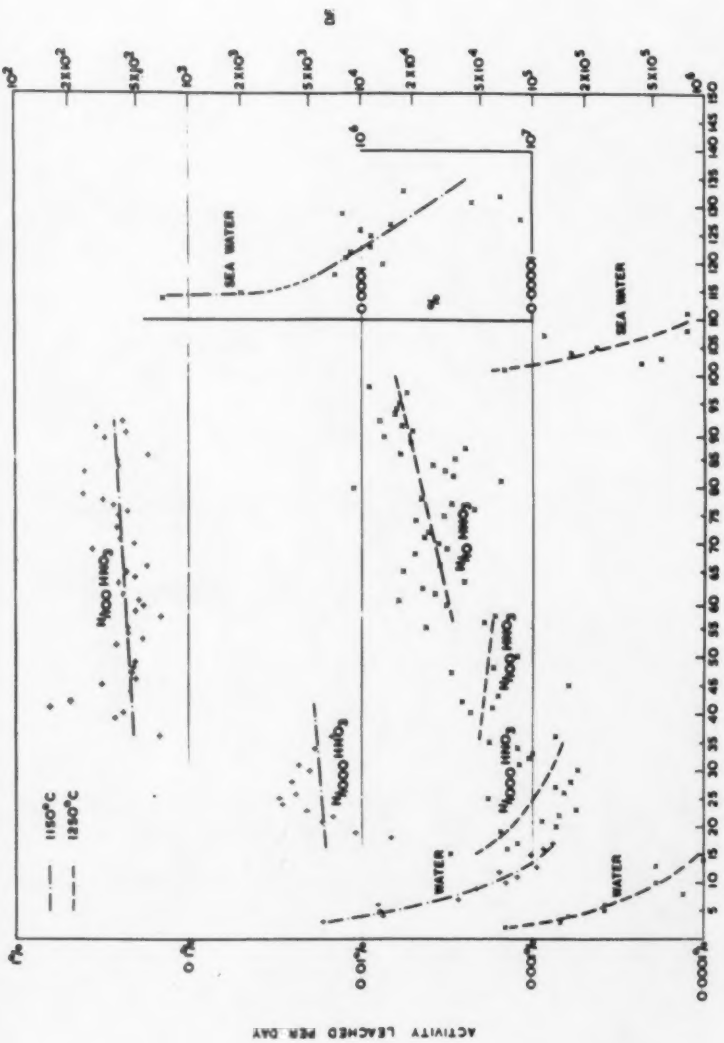


FIG. 7.

PERCENTAGE ACTIVITY LEACHED PER DAY FROM A SAMPLE OF CLAY GRANULES FIRED AT 1150°C AND AT 1250°C FOR 1 HR.

1250°C FOR 1HR.

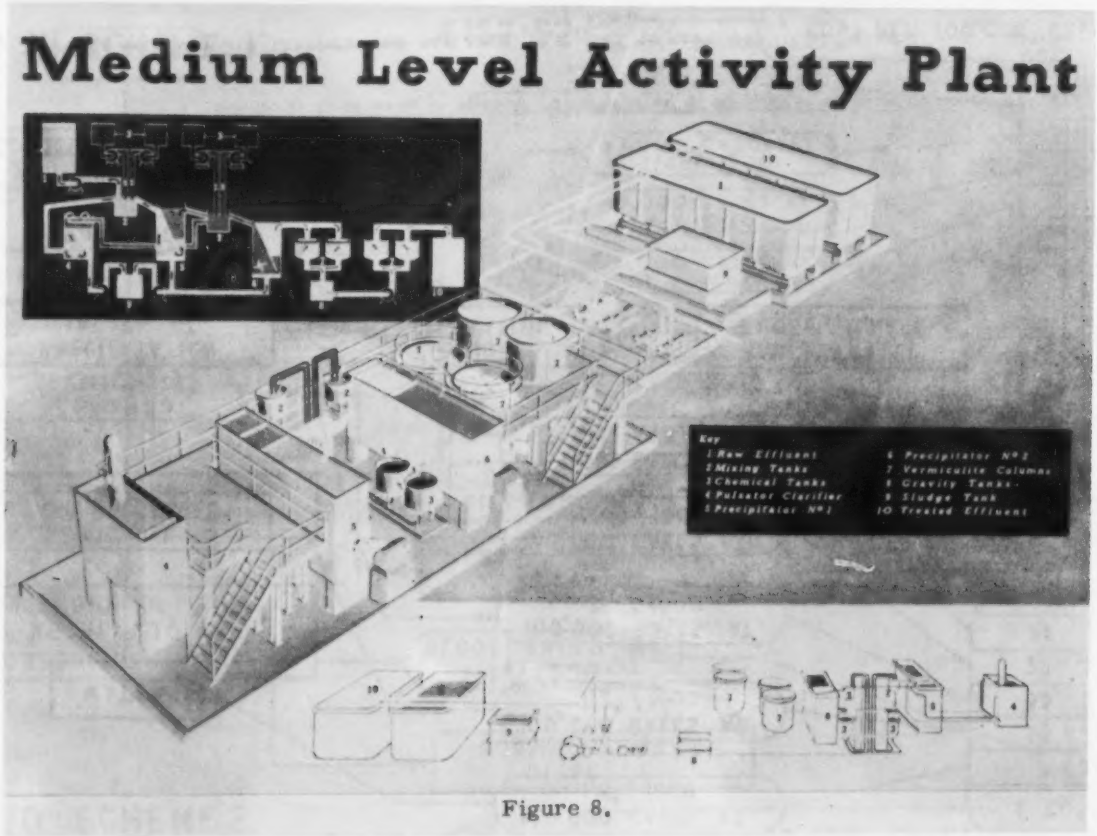


Figure 8.

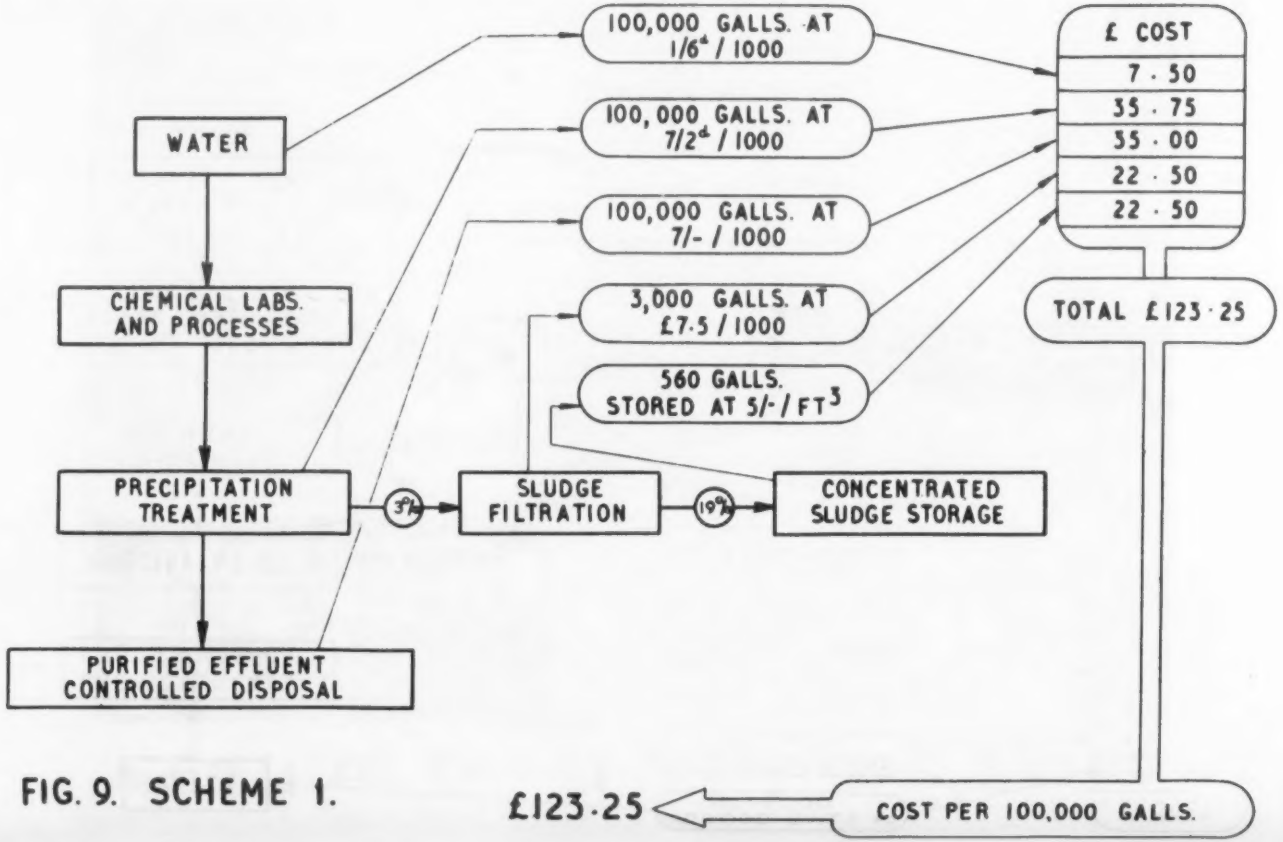


FIG. 9. SCHEME 1.

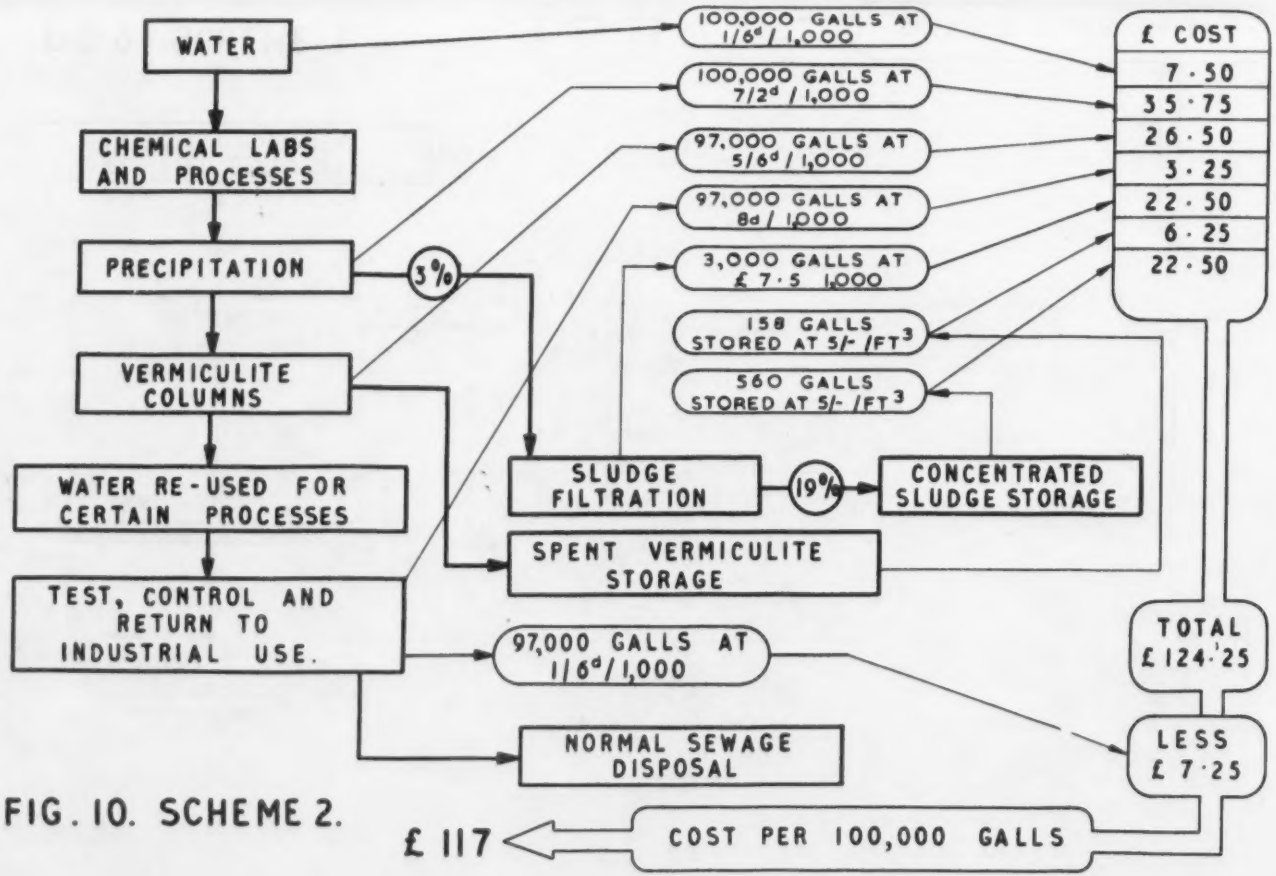


FIG. 10. SCHEME 2.

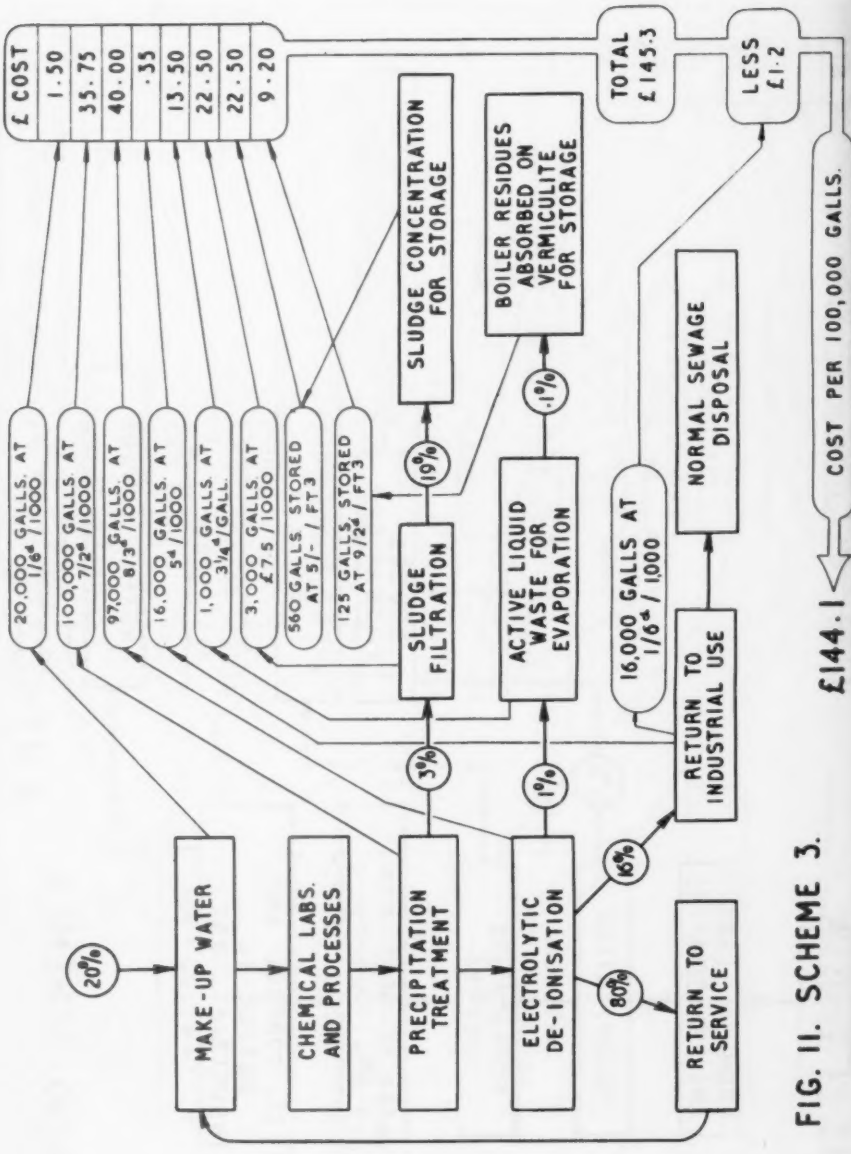


FIG. 11. SCHEME 3.